ENCLOSURE 3

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 AND 50-324/LICENSE NOS. DPR-71 AND DPR-62
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING
REQUEST FOR LICENSE AMENDMENTS - EXTENDED POWER UPRATE
(NRC TAC NOS. MB2700 AND MB2701)

Non-Proprietary Version of Response to Request For Additional Information (RAI) 11

Background

On August 9, 2001 (Serial: BSEP 01-0086), Carolina Power & Light (CP&L) Company requested a revision to the Operating Licenses (OLs) and the Technical Specifications for the Brunswick Steam Electric Plant (BSEP), Units 1 and 2. The proposed license amendments increase the maximum power level authorized by Section 2.C.(1) of OLs DPR-71 and DPR-62 from 2558 megawatts thermal (MWt) to 2923 MWt. Subsequently, on December 10, 2001, the NRC provided an electronic version of a RAI requesting information associated with the Civil & Engineering Mechanics Section's review of the extended power uprate (EPU) amendment request. The responses to this RAI follow.

NRC Question 11-1

In Reference to Section 3.1.1 of Enclosure 3 to the amendment request, you state that EPU evaluations are performed using the existing SRV setpoint tolerance analytical limits as a basis. The in-service surveillance testing of the plant's SRVs has not shown a significant propensity for high setpoint drift greater than 3%. Out of 55 SRV tests, from the "as found" setpoint lift verification tests performed from 1997 to 2001, only four of the SRVs were found to exceed their setpoint by greater than 3%. Confirm whether the BSEP EPU SRV analysis are performed using + 3 percent setpoint tolerance.

Response to Question 11-1

The BSEP EPU analysis/evaluations, which use the Safety/Relief Valve (SRV) opening spring set pressures, were performed using a tolerance of +3%.

NRC Question 11-2

In reference to Section 3.3.2 of Enclosure 3 to the amendment request, you indicate that the effect of EPU was evaluated to ensure that the reactor vessel components continue to comply with the existing structural requirements of the ASME Boiler and Pressure Vessel Code. For the components under consideration, the 1971 Edition of the Code with addenda to and including Summer 1973, which is the construction code of record, was used as the governing Code. You also indicate that if a component underwent a design modification, the governing code for that component was the code used in the stress analysis of the modified component. Provide a summary of the components that were modified and the code editions/code cases (if applicable) other than the code of record that were used for the EPU evaluation.

Response to Question 11-2

The only components that had modifications and also had operating conditions change as a result of EPU are the feedwater nozzles in BSEP Unit 1. The other reactor pressure vessel (RPV) components had no increase in operating conditions due to EPU and were not reanalyzed. For BSEP Unit 1, the feedwater nozzles had safe end replacements and the American Society of Mechanical Engineers (ASME) Code edition used in this analysis is the 1971 Edition including the Addenda through Summer 1973.

NRC Question 11-3

In reference to Section 3.3.2, you indicate that new stresses are determined by scaling the "original" stresses based on the EPU conditions. The analyses were performed for the design, normal, upset, emergency and faulted conditions. Provide a summary discussion of how you arrived to the scaling factors for the EPU at various service conditions. Also, provide an example to illustrate how scale factors were calculated and used in calculating the EPU stress and CUF at the feedwater nozzle blend radius.

Response to Question 11-3

General Electric (GE) has developed a technique to conservatively scale the original stress report stresses to account for changes in the original pressures, temperatures, and nozzle flows as a result of EPU.

Many pressure vessel calculations select the three stress directions of the orthogonal coordinate system such that the shear stress components are zero; the normal stress components are the principal stresses. With this orientation, the pressure stresses are directly proportional to the increase in coolant pressure, and the magnitude of the principal stress resulting from thermal cycling is proportional to the temperature change during a thermal transient. When there are no changes in mechanical loads as a result of the EPU, the new magnitude of the principal stress is:

$$\sigma_{new} = \sigma_p * (P_{new}/P_{old}) + \sigma_t * (\Delta t_{new}/\Delta t_{old}) + \sigma_m$$

Where:

σ_p = Original pressure stress
σ_t = Original thermal stress
σ_m = Original mechanical stress
P_{new} = EPU pressure

P_{old} = Original pressure
Δt_{new} = EPU temperature range
Δt_{old} = Original temperature range

or:

$$\sigma_{new} = \sigma_p * SCF_p + \sigma_t * SCF_t + \sigma_m$$

Where:

SCF_p = Pressure stress scaling factor
SCF_t = Thermal stress scaling factor

Components that experience a change in internal coolant flow during operation have a flow scaling factor, SCF_f. The magnitude of the internal flow changes the convective heat transfer coefficient. The Biot Modulus is used to determine the effect of increased nozzle flows on the nozzle thermal skin stresses. It can be shown that:

$$B_i \ \alpha \ h \ \alpha \ V^{0.8}$$

Where:

B_i = Biot Modulus, hL/k
h = Film convection coefficient
k = Thermal conductivity coefficient
L = Nozzle half wall thickness
V = Flow velocity through nozzle

This relationship allows the flow scaling factor to be determined by the following:

$$SCF_f = (V_{new}/V_{old})^{0.8}$$

When the flow scaling factor is applied, a new thermal SCF is calculated using both the SCF_t and the SCF_f. The new thermal scaling factor, SCF_T, is calculated using the following formula:

$$SCF_T = SCF_t * SCF_f$$

Most stress reports do not separately report the pressure, thermal, and mechanical stresses; therefore, it is not practical to calculate the scaled pressure or scaled thermal stresses. A conservative scaling technique, using the larger of the pressure and temperature scaling factors, is used to scale the entire stress magnitude. If a calculated SCF is less than unity, a SCF = 1.0 is used instead. This method is a conservative alternative to scaling the individual stress components because:

- The largest scaling factor is used for both the pressure and temperature SCF, and

- The mechanical stresses are increased by the SCF even though the design mechanical loads did not increase.

Conditions which generate a stress reduction (i.e., a SCF less than 1.0) are ignored.

The stress scaling technique may be further simplified by applying the SCF to the stress intensity alone, rather than applying the SCF to the principal stress components. A stress intensity, or stress difference, used to compare with the ASME Code allowable values, is determined by selecting the absolute value of the maximum difference between any pair of principal stresses. Consider the following example:

$$\begin{aligned} S_{12,new} &= \sigma_{1,new} - \sigma_{2,new} \\ &= \sigma_{1,old} * SCF - \sigma_{2,old} * SCF \\ &= (\sigma_{1,old} - \sigma_{2,old}) * SCF \\ &= S_{12,old} * SCF \end{aligned}$$

The feedwater nozzle has the highest cumulative usage factor (CUF) at the safe end. The blend radius was not analyzed.

NRC Question 11-4

In reference to Section 3.3.2, you indicate that if there is an increase in annulus pressurization (AP), jet reaction, pipe restraint or fuel lift loads, the changes are considered in the analysis of the components affected due to upset, emergency and faulted conditions. Provide a summary discussion of how these loads are affected by the proposed power uprate. Confirm whether and how these loads are incorporated in the EPU evaluation of the reactor vessel and internal components.

Response to Question 11-4

During preparation of the this response, a question arose as to whether the evaluation of the affect of EPU on internal component stresses fully bounds BSEP's current licensing bases. Resolution of this issue is in progress. CP&L will provide an update to this response when the question is resolved.

NRC Question 11-5

In reference to Section 3.3.2.2, Table 3-3 indicates that the current CUFs of the reactor vessel components reflect the fatigue usage at plant operation of 27 years and 28 years for BSEP Units 1 and 2, respectively, based on the fatigue monitoring system in place at BSEP. The EPU CUFs were calculated using the cycle count from the reactor thermal cycle diagram for the remainder of the 40 years life. Provide a summary describing the fatigue monitoring system at BSEP and a summary of the steps for predicting CUFs using thermal cycle diagram cycle count.

Response to Question 11-5

Brunswick Fatigue Monitoring Program

The Brunswick Fatigue Monitoring Program monitors the five most limiting components of the reactor pressure vessel. These components are 1) Refueling Bellows Support, 2) Reactor Vessel Head Closure Studs, 3) Recirculation Inlet Nozzles, 4) Core Spray Nozzles, and 5) Feedwater Nozzles.

A formal fatigue update for these five components is performed approximately each 10 year period. These updates provide the accumulated fatigue usage based on the accumulated number of cycles for various transients and the severity of the transients as quantified by pressure, temperature, etc.

Interim fatigue usage updates are performed after each refueling outage. These updates consider the fatigue contribution of various operating transients on the five selected components for the preceding operating cycle. These values are added to the previously computed CUF on an ongoing basis.

Procedure for EPU Fatigue Evaluation

The following general procedure describes the standard method used to perform the ASME Code fatigue evaluation for EPU conditions.

1. Determine the applicable SCF for each stress component composing the primary plus secondary plus peak (P + Q + F) stress cycle considered. This includes pressure, temperature, and flow SCFs.

2. Apply the appropriate SCF to the corresponding stress component of the P + Q + F stress cycle.

3. Determine the alternating stress intensity, then calculate and apply the correct fatigue strength reduction factor, $K_{e,new}$ and elastic modulus correction factor, E_c/E_a. Both of these factors are described in the Code. Because the value for n and m vary by material and are included in the Code, they are not listed here. The uprated $S_{alt,new}$ takes the following form:

$$S_{alt,new} = \frac{1}{2} * K_{e,new} * (E_c/E_a) * S_{P+Q+F,new}$$

Where:

$$K_{e,new} = \text{Simplified elastic-plastic factor}$$
$$= 1.0, \text{ for } S_{n,new} \leq 3S_m$$

$$= 1.0 + [(1-n)/n(m-1)] * [(S_{n,new}/3S_m) - 1],$$
$$\text{for } 3S_m < S_{n,new} < 3mS_m$$
$$= 1/n, \text{ for } S_{n,new} \geq 3mS_m$$

4. Using $S_{alt,new}$, determine the Code allowable number of cycles from the fatigue curve appropriate to the material of the component under consideration.

5. Repeat Steps 2-4 for each peak stress intensity corresponding to a group of cycles considered in the fatigue analysis.

6. The cumulative fatigue usage factor may now be determined from the following formula:

$$U = u_1 + u_2 + \ldots + u_x$$

Where:

u_x = n_x/N_x, is the incremental fatigue usage factor
n_x = expected number of lifetime cycles experienced by the component
N_x = Code allowable number of cycles determined in Step 4.

7. Compare the cumulative usage factor to the Code allowable upper limit, $U < 1.0$.

This section is not intended as a replacement for the requirements or methods described in the ASME Code; the discussion is provided only to describe the standard GE methodology.

The Brunswick fatigue usage calculation is a summation of the usage measured from the fatigue monitoring system for the first 27 and 28 years for BSEP Units 1 and 2, respectively, and the final 13 (or 12 years) based on the above described procedure.

NRC Question 11-6

In reference to Section 3.3.2.2, Table 3-3 indicates that the reactor vessel components do not experience a change in stresses due to EPU. Provide a summary of loads and design transients considered in the evaluation and the allowable stress limits for each evaluated component. Confirm whether the loads used in the evaluation of the reactor vessel components are not affected by the EPU.

Response to Question 11-6

Table 3-3 shows that the feedwater nozzles, for both BSEP Units 1 and 2, experience an increase in stress due to EPU conditions. A power uprate was performed in 1995. The only changes in operating conditions since that time, that affected the RPV components with fatigue usage values above 0.5, are shown below:

- Feedwater Nozzle: Feedwater inlet temperature increases from 427°F to 431.4°F (for Unit 1) and 419°F to 431.4°F (for Unit 2).

- Feedwater Nozzle: The 100% rated flow/nozzle increases from 6,800 gpm to 7,763 gpm (for Unit 1) and from 6,380 gpm to 7,763 gpm (for Unit 2).

The other components listed in Table 3-3 have the following operating conditions, which remain unchanged since the 1995 power uprate:

1. Region A: Pressure remains at 1035 psig (originally was 1000 psig).

2. Regions A, B, and C: Operating temperature remains at 551°F (originally was 546°F).

3. In the "scram" transient, the following region A pressures are as follows: 1215 psig (originally 1180 psig), 910 psig (originally 875 psig), 1160 psig (originally 1125 psig), 1035 psig (originally 1000 psig), 700 psig (originally 665 psig), 965 psig (originally 930 psig).

4. In regions B and C the operating temperatures remain: 534°F (originally 527°F), 551°F (originally 543°F), 524°F (originally 517°F).

5. Recirculation Inlet Nozzle 100% flow remains at 9560 gpm (originally 8700 gpm).

According to the ASME Code, structural adequacy is demonstrated if the maximum primary plus secondary stress intensity range (S_n) at a location on the component is less than $3S_m$ of the material at the location examined. If this limit is exceeded, a simplified elastic-plastic analysis may be used to demonstrate structural adequacy.

NRC Question 11-7

In Section 3.3.3, you state that the original acoustic loads on the reactor internal components, following a postulated recirculation line break, were also updated in accordance with current methodology. Provide a summary of the methodology and assumptions used in calculating the acoustic loads and provide an example to illustrate how the acoustic loads and flow induced loads were calculated, at the critical locations (i.e., shroud), due to recirculation line break for the EPU condition.

Response to Question 11-7

The updated methodology for calculating the loads on reactor internal components for a postulated recirculation line break is based on the studies performed in support of reactor shroud cracking concerns.

Acoustic Load Methodology Summary

The acoustic loads, which occur just after an instantaneous pipe break, are due to a decompression wave in the vessel, with reflections from and attenuations by jet pumps, and the subsequent reflection of the wave against itself on the far side of the reactor vessel from the pipe break.

[Redacted]

Flow Induced Load Methodology Summary

[Redacted]

Table 10.7-1 [Redacted]					

Table 10.7-2 [Redacted]			

Table 10.7-3 [Redacted]			

NRC Question 11-8

In Section 3.3.3, you also state that minor components such as steam dryer/separator rods, in-core guide tube braces, and jet pump sensing lines are not affected by the EPU and were not evaluated. In light of an BWR power uprate, the modification was necessary for the in-core guide tube braces because the in-core guide tube are very sensitive and susceptible to vibration from the recirculation pump vane passing frequency (VPF) which will slightly increase for the EPU. Provide the technical basis for your conclusion.

Response to Question 11-8

Section 3.3.3 addresses the effects of reactor internal pressure differences (RIPD) due to EPU. Section 3.3.3 states that the RIPD does not affect minor components such as steam dryer/separator guide rods, in-core guide tube braces, and jet pump sensing lines.

The vibration effects are addressed in Section 3.3.5. The in-core guide tubes vibrations are mainly a function of core flow, which does not change during EPU. The in-core guide tubes are not susceptible to vibration from the recirculation pump VPF. The in-core guide tube braces were not modified due to EPU at any plant.

The only components that have been susceptible to recirculation pump VPF, in some other plants, are the jet pump riser brace leaves and the jet pump sensing lines (JPSL). After discovery of the initial JPSL failures in 1982, GE performed extensive analysis and tests at its test facilities to determine the cause of these failures. The results of the analysis and tests showed conclusively that the failures were due to high amplitude vibration resulting from resonance of the JPSL second natural frequency with the VPF of the recirculation pumps.

The jet pump riser brace leaves and the JPSLs were evaluated for BSEP at EPU conditions. Detailed finite element dynamic analyses of the JPSLs have shown that the recirculation pump VPFs are well separated from the natural frequencies of interest of the JPSLs. Therefore, there is no concern for resonance vibration of the jet pump riser brace leaves and JPSLs with the VPF of the recirculation pumps.

NRC Question 11-9

In Section 3.3.4, you indicate that for components experiencing increased loads due to EPU, the existing stresses are scaled-up in proportion to the loads, and the combined stresses and fatigue usage factors were compared to the code allowables for the various service conditions. Provide a summary describing how you arrived to the scaling factors for the EPU at various service conditions. Also, provide an example to illustrate how scale factors were calculated and the calculation of the EPU stress and CUF at the feedwater nozzle and at the critical location on the shroud.

Response to Question 11-9

To select the scaling factor, the loads affected by the EPU (i.e., pressure, flow and temperature, etc.) on a component are compared between EPU and current licensed thermal power (CLTP) conditions. Then, the maximum scaling factor is selected. The critical response is scaled up with this maximum factor and compared to the allowable for qualification. For an example of the scaling process, refer to Response to Question 11-10.

In the case of the shroud, the actual EPU loads were used in reconciling the maximum stress, consistent with the existing design basis. As a result of EPU, the only loads affected are the differential pressures. The shroud stress determined for the Upset condition is provided, as an example. The applicable loads, for the shroud in the upset condition, are the differential pressures across the shroud head (i.e., 13.49 psi) and the core plate (i.e., 30.85 psi), dead weight, and operating basis earthquake (OBE) seismic loads. The membrane and bending stresses in the shroud, as a result of these loads, are calculated and combined which resulted in a stress of 3938 psi. The CLTP stress in the existing design basis was 4116 psi. The allowable stress consistent with the design basis is 14,500 psi.

An example of the scaling factors used in calculating the EPU stress and CUF is provided for the BSEP Unit 1 feedwater nozzles. The methods used are the same as those used in the 1995 power uprate; however, the scaling factors are calculated based on original vessel conditions rather than those used in the 1995 power uprate. Also, the stresses are scaled from the original stresses rather than those of the 1995 power uprate. For the feedwater nozzle fatigue analysis, there were 10 stress event combinations that each result in incremental fatigue. The summation of these 10 incremental fatigue usages results in the final fatigue usage for the feedwater nozzle. For this example, only one load combination (i.e., loss of feedwater pumps event to the turbine roll event) is considered and the resulting incremental fatigue usage calculated.

Original Analysis Summary:

During the loss of feedwater pumps event, the stress intensities are:

$$S_{P+Q} = 77.5 \text{ ksi (primary + secondary)}$$
$$S_{P+Q+F} = 87.0 \text{ ksi (primary + secondary + peak)}$$

During the turbine roll event, the stress intensities are:

$$S_{P+Q} = -36.0 \text{ ksi}$$
$$S_{P+Q+F} = -40.2 \text{ ksi}$$

Nominal (i.e., primary plus secondary) stress intensity: $S_n = 77.5 - (-36.0) = 113.5$ ksi

Peak stress intensity range: $S_p = 87.0 - (-40.2) = 127.2$ ksi

Fatigue strength reduction factor:
$$\begin{aligned} K_e &= 1.0 + [(1-n)/n(m-1)] \times [(S_n/3S_m) - 1] \\ &= 1.0 + [(1-.3)/.3(1.7-1)] \times [(113.5/69.9) - 1] \\ &= 3.08 \end{aligned}$$

where, $n = .3$, $m=1.7$, and $3S_m = 69.9$ ksi

Alternating (peak) stress:
$$\begin{aligned} S_{alt} &= 1/2 \times K_e \times (E_c/E_a) \times S_p \\ &=1/2 \times 3.08 \times (26/29.5) \times 127.2 \\ &= 173 \text{ ksi} \end{aligned}$$

where, E_c/E_a = elastic modulus correction factor.

The safe end material is SA-508. The number of allowable cycles for an alternating stress of 173 ksi is 246 (i.e., $N_{all} = 246$). The actual number of cycles that the nozzle will see is 120. The incremental fatigue, $u_x = 120/246 = 0.4880$

Scaling Factor Calculation for EPU Condition:

During the loss of feedwater pumps event, the original RPV temperatures and feedwater flow rates are as follows:

Initial temperature, $T_{i,old} = 546°F$
Final temperature, $T_{f,old} = 40°F$
Flow rate, $V_{old} = 6500$ gpm

During the loss of feedwater pumps event, the EPU temperatures and flow rates are as follows:

Initial temperature, $T_{i,new} = 551°F$
Final temperature, $T_{f,new} = 40°F$
Flow rate, $V_{new} = 7763$ gpm

Temperature scale factor = $(T_{i,new} - T_{f,new})/(T_{i,old} - T_{f,old}) = (551 - 40)/(546 - 40) = 1.0099$

Flow scale factor = 1.0182

The flow scaling factor for the BSEP Unit 1 feedwater nozzles was calculated using a formulation based on the Biot Modulus. The heat transfer coefficient, however, is scaled by the flow to the power of 0.8. The steps in this formulation are detailed below:

a) Calculate a new heat transfer film coefficient, h, based on the increase in flow.

$h_{orig} = 799$ BTU/hr-ft^2-°F
$h_{EPU} = h_{orig} \times (\text{new flow/old flow})^{0.8} = 799 \times (7763/6500)^{0.8} = 921$ BTU/hr-ft^2-°F

b) Calculate a new thermal conductivity, k, based on the feedwater nozzle temperature change.

Original operating conditions ($T_{FW} = 420°F$): k = 10.10 BTU/hr-ft-°F
EPU operating conditions ($T_{FW} = 431.4°F$): k = 10.16 BTU/hr-ft-°F

c) Calculate a new diffusivity, TD, based on the feedwater nozzle temperature change.

Original operating conditions ($T_{FW} = 420°F$): TD = 0.1555 ft^2/hr
EPU operating conditions ($T_{FW} = 431.4°F$): TD = 0.1559 ft^2/hr

d) Calculate a new Biot Modulus.

Biot Modulus = h × l/k (dimensionless)

Where: h, heat transfer coefficient = 921 BTU/hr-ft^2-°F

l, ½ material thickness = 0.052 ft

k, thermal conductivity = 10.16 BTU/hr-ft-°F

EPU Biot Modulus = (921 × 0.052)/10.16 = 4.72

e) Calculate the Fourier Number.

Fourier Number = (TD × t)/l^2

Where: TD, diffusivity = 0.1555 ft^2/hr (original) = 0.1559 ft^2/hr (EPU)

t, time (function of transient step function used) = 0.005 hr

l, slab thickness = 0.052 ft.

Fourier Number (original) = (0.1555 × 0.005)/0.052^2 = 0.2867

Fourier Number (EPU) = (0.1559 × 0.005)/0.052^2 = 0.2874

f) Calculate log Fourier Number.

Log Fourier Number (original) = -0.5426

Log Fourier Number (EPU) = -0.5416

g) Look up L1 (i.e., Reference: D.R. McNeill and J.E. Brock, "Charts for Transient Temperatures in Pipes," Heating/Piping/Air Conditioning, November 1971) for a step function.

L1(original) = 0.55

L1(EPU) = 0.56

h) Calculate the flow scaling factor, SCF_f.

SCF_f = L1(EPU)/L1(original) = 0.56/0.55 = 1.0182

i) Calculate the combined thermal scale factor, SCF_T.

Thermal scale factor, $\mathbf{SCF_T}$ = $SCF_t \times SCF_f$ = 1.0099 × 1.0182 = **1.0282**

EPU Analysis Summary:

During the loss of feedwater pumps event:

Temperature scale factor, SCF_t = 1.0099 (i.e., calculation shown above)

Flow scale factor, SCF_f = 1.0182 (i.e., calculation shown above)

Thermal scale factor, $SCF_T = SCF_t \times SCF_f = 1.0099 \times 1.0182 = 1.0282$

$S_{P+Q} = 77.5 \times 1.0282 = 79.7$ ksi

$S_{P+Q+F} = 87.0 \times 1.0282 = 89.5$ ksi

During the turbine roll event:

Temperature scale factor, $SCF_t = 1.0$ (i.e., no change in temperature due to EPU)

Flow scale factor, $SCF_f = 1.0154$ (i.e., calculation not shown, based on a different transient)

Thermal scale factor, $SCF_T = SCF_t \times SCF_f = 1.0 \times 1.0154 = 1.0154$

$S_{P+Q} = -36.0 \times 1.0154 = -36.6$ ksi

$S_{P+Q+F} = -40.2 \times 1.0154 = -40.8$ ksi

Nominal primary + secondary stress intensity range: $S_n = 79.7 - (-36.6) = 116.3$ ksi

Range of peak stress intensity: $S_p = 89.5 - (-40.8) = 130.3$ ksi

Fatigue strength reduction factor:

$$\begin{aligned} K_e &= 1.0 + [(1\text{-}n)/n(m\text{-}1)] \times [(S_n/3S_m) - 1] \\ &= 1.0 + [(1\text{-}.3)/.3(1.7\text{-}1)] \times [(116.3/69.9) - 1] \\ &= 3.21 \end{aligned}$$

where, $n = .3$, $m=1.7$, and $3S_m = 69.9$ ksi

Alternating stress:

$$\begin{aligned} S_{alt} &= 1/2 \times K_e \times (E_c/E_a) \times S_p \\ &= 1/2 \times 3.21 \times (26/29.5) \times 130.3 \\ &= 184.3 \text{ ksi} \end{aligned}$$

where, E_c/E_a = elastic modulus correction factor

The safe end material is SA-508. The number of allowable cycles for an alternating stress of 184.3 ksi is 202 (i.e., $N_{all} = 202$). The actual number of cycles that the nozzle will see is 120. The incremental fatigue, $u_x = 120/202 = 0.5931$. This is the usage due to this event combination only. It is not the total usage for the feedwater nozzle. Nine other event combinations were analyzed and their fatigue contributions must be added to this usage to come up with the total fatigue usage for the feedwater nozzles.

NRC Question 11-10

You indicated that the steam dryer pressure differentials at EPU conditions are slightly larger than the design basis. Consistent with the design basis, the EPU RTP stresses are reconciled and the analysis shows that the revised stresses are within the allowables. Provide a summary of comparison for the design basis and the EPU condition with respect to design loads and load combinations for various service conditions. Also, provide the calculated maximum stresses for the critical locations of the dryer structure including supporting bracket and the welds.

Response to Question 11-10

As the general process for reconciling the load changes for the EPU, first the loads affected by the EPU are reviewed. If the loads are unaffected or smaller than the design basis, no further evaluation is required or performed. However, if a load increases beyond the design basis, such increase is reconciled either by conservatively scaling the original stresses, or by calculating the combined stress based on the various loads applicable to the service condition. In this manner, the design basis load combinations are maintained for the EPU. Only critical stress locations are reconciled. In the case of the steam dryer, the scaling method is conservatively used for the Upset condition, as demonstrated below as an example.

The only load affecting the steam dryer is the pressure load. The differential pressures across the components affecting the steam dryer, based on existing/design basis and EPU, are compared for Upset condition. The Upset Condition Differential Pressure is 0.52 psi for design basis and 0.7 psi for EPU, leading to a ratio of EPU/design basis of 1.346. Therefore, the design basis Upset condition combined stresses for the critical dryer components, as in the design basis, are scaled up by a factor of 1.346, as shown below:

Location	Original Stress (psi)	Scale Factor	Scaled EPU Stress (psi)	Allowable Stress (psi)
Hood Stress (Pm)	358	1.346	482	16,900
Hood Stress (Pm + Pb)	1,959	1.346	2,637	25,350
Support Ring	14,099	1.346	18,979	25,350

NRC Question 11-11

In Section 3.3.5, you provide a list of components (including steam dryer) that were evaluated for the flow induced vibration. You also indicate that during EPU operation, the components in the upper zone of the reactor, such as the steam separators and dryers, are mostly affected by the increased steam flow. Provide recorded or testing data and a summary of the evaluation with regard to the flow induced vibration affecting steam dryers. Discuss the potential for flow-induced vibration of the steam dryers due to various mechanisms, including, in particular, the fluid-elastic instability in the steam separators and dryers at the proposed power level. If the details of the analysis and the results are documented in a report, submit the report for staff review. In light of the discussion in GE SIL No. 474 and BWRVIP-06 report, discuss how you can ensure that the steam dryer and dryer supporting bracket will maintain its structural integrity during the EPU operation.

Response to Question 11-11

The steam dryer has no safety function. The sole function of the steam dryer is to remove moisture from the steam in order to minimize erosion of the piping and turbine and to improve the turbine efficiency. BWRVIP-06, which was endorsed by the NRC, also states that the dryer is non-safety related and failure of a dryer component may cause an operability concern but has no safety impact. Hence the dryer was not instrumented during startup testing and no measured vibration data is available for the prototype plant.

The design criteria for the steam dryer is that the structural integrity of the dryer is maintained when subjected to a steam line break occurring beyond the main steam isolation valves. Since the reactor vessel steam dome pressure is not changed under EPU conditions, steam dryer structural integrity evaluations, performed for a steam line break for the current rated thermal power, are applicable to EPU conditions.

The operational histories of dryers in other boiling water reactor (BWR) plants were also studied to determine if there were any flow-induced vibration-related problems in the dryer. Only drain channel cracks at steady state conditions and outer bank hood damage due to turbine stop valve (TSV) closure were found that were attributable to vibration effects. Drain channel cracking, discussed in GE SIL 474, has occurred during normal operation and is usually repaired after detection. The outer bank hood (i.e., adjacent to the steam outlet nozzles) at BSEP is four times thicker than at the plant where the damage occurred, while the TSV closure time is identical. Hence it is expected that the outer bank hood can withstand the transient. While instances of drain channel cracking and hood cracking have occurred at operating plants, it is an operational issue only, relating to proper drying of the steam before it leaves the dryer. No structural integrity problems have been observed with these cracks. The dryers are visually inspected during removal in each refueling outage and any significant cracking can be repaired.

The basic mechanisms of flow induced vibration (FIV) in reactor internal components are due to (a) cross flow, (b) turbulent parallel flow, (c) forced vibration due to recirculation pump pulsation at the VPF, and (d) motion dependent force excitation. Of these mechanisms, only turbulence and forced vibration due to pump pulsation at the VPF have signifcant effects on BWR internal components. The effects of these depend on the location of the component and the flow. The characteristics of the responses have been determined by testing performed at development test facilities and at prototype reactors during initial startup. Based on years of testing, the only BWR reactor internal component, which may be subjected to fluid-elastic instability, is the jet pump. Thus, steam separator and dryer vibration due to fluid -elastic instability effects are not anticipated.

With respect to the dryer support bracket, BWRVIP-06 states:

> Failure of multiple dryer support brackets, a very unlikely scenario, could result in drop of the dryer onto the moisture separator. This would cause a detectable change in steam flow, and thus power, and would prompt a shut down. A dryer drop to the moisture separator would result in loads being imposed on the shroud head, shroud, and shroud support, all structurally robust components. Therefore, control rod insertion and safe shutdown would be achieved.

The dropping of the dryer onto the separator due to failure of dryer support brackets has never happened in any BWR. Thus, failure of steam dryer support brackets is not a safety issue.

NRC Question 11-12

In reference to Section 3.5, provide a discussion of the methodology and assumptions used for evaluating the reactor coolant pressure boundary piping (RCPB) systems for the proposed power uprate. If scale factors are used in calculating the stresses at the EPU condition, provide a summary describing how you have arrived to the scaling factors for the EPU at various service conditions and an example to illustrate how scale factor magnitudes were calculated and the calculation of the EPU stress and CUF (i.e., for the main steam and feedwater piping). Also, provide the calculated maximum stresses and fatigue usage factors for both the current design basis and the EPU conditions at critical locations on the RCPB piping systems, allowable stress limits, and the code and code edition used in the evaluation for the power uprate. If different from the code of record, justify and reconcile the differences. Were the analytical computer codes used in the evaluation different from those used in the original design-basis analysis? If so, identify the new codes used and provide your justification for their use by specifying how were these codes benchmarked for such applications.

Response to Question 11-12

Evaluation methodologies for the piping and associated structures such as nozzles, supports etc., are described in ELTR1, Section 5.5.2 and 5.10.10, and ELTR2, Supplement 1, Volume 1, Section 4.8. The EPU parameters of piping systems (i.e., pressure, temperature, and flow) were compared with the corresponding pre-EPU values to determine the increases in temperature, pressure, and flow due to EPU conditions. The multiplying factors were then used to determine the percentage increases in applicable United States American Standard (USAS) B31.1 Code stresses, displacements, and pipe interface component loads, including supports, as a function of percentage increase in pressure, temperature, and flow. The percentage increases due to pressure, thermal, and flow for EPU were applied to the highest calculated stresses, displacements at applicable piping system node points to conservatively determine the maximum EPU calculated stresses, and displacements. This approach is conservative since the factor is applied to the combined loads and the EPU does not affect weight and dynamic loads (e.g., seismic load). The factors were also applied to nozzle loads, support loads, penetration loads, valves, pumps, heat exchangers, nozzles, and anchors so that these components could be evaluated for acceptability, where required.

An evaluation process has been developed by GE that precludes the necessity for reanalysis of piping impacted by EPU. Instead of reanalysis, the existing piping design basis is evaluated using a set of unique parametric curves. These curves allow the prediction of the ASME Section III and B31.1 Power Piping Code equations.

The parametric study was performed to evaluate the EPU effect on Nuclear Steam Supply System (NSSS) and Balance of Plant (BOP) piping for BWR 4, 5 and 6 nuclear power plants due to temperature, pressure, and flow change. The parametric study focuses on plant piping that may be impacted by EPU increases in pressure, temperature, and flow. Applying a range of percentage increases to the piping system pressure, temperature, and flows and evaluating the effects on Code stresses, pipes support loads are determined based on these increases.

The following tables provide examples of the above approach which address BSEP Units 1 and 2 FW pipe stresses, for different USAS B31.1 Code equations. These tables provide stress node points, maximum current design values, EPU values after multiplying a factor, and ratio to allowable. The USAS B31.1 Power Piping Code does not require the calculation of the fatigue usage factor; hence, it was not evaluated for feedwater piping system. The recirculation piping is not impacted since BSEP is undergoing a constant pressure power uprate and flow change is insignificant.

[Redacted - Four Tables]

The following table summarizes the piping systems that are potentially impacted by EPU.

System	Description
Main Steam A	Main Steam Piping System
Main Steam B	Main Steam Piping System
Main Steam C	Main Steam Piping System
Main Steam D	Main Steam Piping System
Reactor Core Isolation Cooling (RCIC)	Inside Containment Steam Side
High Pressure Coolant Injection (HPCI)	Inside Containment Steam Side
Safety Relief Valve Drain Lines (SRVDLs)	Up to First Anchor
Reactor Pressure Vessel Head Vent	RPV Head Vent Piping
Main Steam Isolation Valve Drain Lines (MSIVDLs)	Inside Containment Main Steam Drain Lines Piping
Feedwater	Feedwater Pipins Systems Loop A and Loop B
Core Spray	Core Spray Piping

Since this is a constant pressure power uprate, the main steam and connected piping (i.e., RCIC, HPCI, SRVDLs, MSIV Drain Lines, and RPV Head Vent) are not affected by pressure and temperature changes. The EPU will result in an increase in steam flow of approximately 15% over current conditions. The consideration of steam flow transients on main steam piping, other than safety relief valve discharge loads, is beyond the BSEP design basis. For BSEP, the consideration of these transient loads is specifically exempted by the Standard Review Plan (SRP), Section 10.3, "Main Steam Supply System," Revision 3, Subsection V. Core Spray system piping is not operating during normal plant operation; hence, it is not affected due to EPU.

The original piping design code for BSEP is USAS B31.1.0 - 1967. As noted in the attachments, some run of record pipe stress calculations were performed using ANSI B31.1 - 1973, which was previously reconciled with the original code of construction. No changes in the code, or code editions, used in the current run of record calculations were made as a result of the EPU evaluations.

No new computer codes were used, or new assumptions introduced, for this evaluation.

NRC Question 11-13

Provide a summary of your evaluation of pipe supports, nozzles, penetrations, guides, valves, pumps, heat exchangers and anchors at the power uprate condition. The evaluation should include the methodology, assumptions, and results of the evaluation for the critical piping systems affected by the proposed power uprate. If scale factors are used in calculating the stresses at the EPU condition, provide a summary describing how you have arrived to the scaling factors for the EPU at various service conditions and an example to illustrate how scale factor magnitudes were calculated and the calculation of the EPU stress and CUF (i.e., for the main steam and feedwater piping). Were the analytical computer codes used in the evaluation different from those used in the original design-basis analysis. If so, identify the new codes and provide your justification for their use by specifying how were these codes benchmarked for such applications.

Response to Question 11-13

The evaluation methodology, assumptions and scaling factors, etc. are described in Response to NRC Question 11-12. A summary of evaluation results of feedwater pipe supports, and nozzles is provided in the following tables.

Description: Evaluation of pipe support loads for Feedwater Piping - Unit 1- Loop A

Feedwater inside Containment	Type of Supports	Total Number of Supports*	Support Acceptable for EPU	Remark
Supports for Unit 1 Loop A	Rigid	None	NA	
	Snubbers	16	YES	Note 1
	Constant Spring Hangers	7	YES	Note 2

Note 1: Snubbers are not active during the normal operation and are not affected by small percentage of EPU temperature increase. Therefore, the snubber deadbands are unaffected.

Note 2: Constant spring hangers are installed to support weight loads only. Since, weight loads are not changed due to EPU, no evaluation of constant spring hangers due to EPU is necessary.

[Redacted - Two Tables]

Description: Evaluation of Pipe Support Loads for Feedwater Piping - Loop B

Feedwater inside Containment	Type of Supports	Total Number of Supports*	Support Acceptable for EPU	Remark
Supports for Unit 1 Loop B	Rigid	None	NA	
	Snubbers	16	YES	Note 1
	Constant Spring Hangers	7	YES	Note 2

Note 1: Snubbers are not active during the normal operation and are not affected by small percentage of EPU temperature increase. Therefore, the snubber deadbands are unaffected.

Note 2: Constant spring hangers are installed to support weight loads only. Since, weight loads are not changed due to EPU, no evaluation of constant spring hangers due to EPU is necessary.

[Redacted - Two Tables]

Description: Evaluation of pipe support loads for Feedwater Piping - Unit 2- Loop A

Feedwater inside Containment	Type of Supports	Total Number of Supports *	Support Acceptable for EPU	Remark
Supports for Unit 2 Loop A	Rigid	None	NA	
	Snubbers	16	YES	Note 1
	Constant Spring Hangers	7	YES	Note 2

Note 1: Snubbers are not active during the normal operation and are not affected by small percentage of EPU temperature increase. Therefore, the snubber deadbands are unaffected.

Note 2: Constant spring hangers are installed to support weight loads only. Since, weight loads are not changed due to EPU, no evaluation of constant spring hangers due to EPU is necessary.

[Redacted - Two Tables]

Description: Evaluation of pipe support loads for Feedwater Piping - Unit 2- Loop B

Feedwater inside Containment	Type of Supports	Total Number of Supports	Support Acceptable for EPU	Remark
Supports for Unit 2 Loop B	Rigid	None	NA	
	Snubbes	16	YES	Note 1
	Constant Spring Hangers	7	YES	Note 2

Note 1: Snubbers are not active during the normal operations and are not affected by small percentage of EPU temperature increase. Therefore, the snubber deadbands are unaffected.

Note 2: Constant spring hangers are installed to support weight loads only. Since, weight loads are not changed due to EPU, no evaluation of constant spring hangers due to EPU is necessary.

[Redacted - Two Tables]

NRC Question 11-14

In Section 3.5.5, you indicate that the MS and FW piping will experience increased vibration levels, approximately proportional to the square of the flow velocities. For the proposed power uprate, the flow rates and flow velocities will increase by more than 20 percent of the flow rate at the original rated thermal power for the MS and FW piping systems. Provide an evaluation of the cumulative fatigue usage factor (in addition to the startup and shutdown cycles), and the potential for flow-induced vibration in the MS and FW piping (during the normal and upset operations), and in the heat exchangers following the power uprate.

Response to Question 11-14

The steady state flow-induced vibration maximum stress levels of the main steam and feedwater piping must remain below the endurance limit of the piping material. This is because many cycles of vibration will be encountered over the remaining design life of the plant. The design fatigue endurance limit for steady state alternating stresses for vibration of 7,690 psi (i.e., zero to peak) for carbon steel piping materials is well below the mean value of the actual endurance limit stress for this material. These fatigue design endurance limits were taken from ASME Section III Pressure Vessel and Piping Code and the ASME/American National Standards Institute (ANSI), OM Part 3, 1987 Edition with 1988 Addenda, for stainless steel and carbon steel materials, respectively.

The potential for FIV of the main steam and feedwater piping due to various FIV mechanisms, such as a fluid-elastic instability, is possible. However, it is not possible to analytically predict which FIV mechanism, if any, may occur within the main steam or feedwater piping at the new and higher main steam and feedwater flow rates associated with the EPU flow conditions. For this reason, BSEP will perform a startup piping vibration test program, during power ascension to the new EPU conditions, as stated in Section 10.4.3 of the PUSAR. These startup tests are intended to show that the steady state main steam and feedwater piping FIV levels, at the new EPU flow conditions, are below the fatigue endurance limit of the piping material.

Startup, shutdown, normal, and upset conditions, or transient vibration cycles associated with the main steam and feedwater piping, are assessed in the piping evaluation report prepared for the planned EPU at the uprated flow conditions.

Heat exchangers in the main steam line piping system and feedwater piping systems, such as the condenser and the feedwater heaters, were assessed at the higher steam and FW flow rates. The condensers are considered adequately protected against tube vibration damage at the EPU conditions. The performance of the feedwater heaters will be monitored for indications of unacceptable vibration during the EPU power ascension program.

If the main steam and feedwater piping vibration levels measured in the planned startup piping vibration test are below the acceptance criteria, then the FIV levels are acceptable and below the fatigue endurance limit of the piping material, independent of the FIV mechanism occurring. Thus, there would be no change in the cumulative fatigue usage factor.

NRC Question 11-15

In Section 10.4.3, "Main Steam Line and Feedwater Piping Flow Induced Vibration Testing," you indicate that the piping vibration stress level for these two piping systems must stay below certain criteria. The allowable alternating vibration stress levels, quantified in Section III of the ASME code, will be used to establish an acceptance criterion for each vibration sensor used for monitoring this piping vibration. Provide a summary describing the acceptance criteria including the allowable vibration stress limits for carbon steel and stainless steel pipes. Confirm whether or not you commit to and meet the provision for testing of the ASME OM-3 Code, "Requirements for Pre-operational and Initial Start-Up Vibration Testing of Nuclear Power Plant Piping Systems."

Response to Question 11-15

The main steam line and feedwater piping flow induced vibration testing uses ASME OM Part 3, 1987 Edition with 1988 Addenda as a guide. The reason for using ASME OM-3 as a guide is the fact that the design basis for the main steam and feedwater piping is the ANSI B31.1 Power Piping Code. The stress criterion in ASME OM Part 3 is based on the ASME Code Section III. Thus, the acceptance criterion for the vibration testing of the main steam and feedwater piping has been modified, such that the stress indices have been replaced with stress intensification factors, consistent with ANSI B31.1.

The main steam and feedwater piping of concern is constructed of carbon steel. The maximum allowable stress due to steady state vibration for the carbon steel pipe is 7,690 psi (i.e., 0.8 Sel/1.3, where Sel = endurance limit from Figure I-9.1 of Section III of the ASME Code). The allowable stress is consistent with ASME OM Part 3 and the testing meets the intent of ASME OM Part 3.

NRC Question 11-16

In Section 10.4.3, you state that vibration data will be collected at 100% of the current licensed thermal power (CLTP). For this EPU, the maximum power level at which data will be taken is at 100% EPU RTP. The measured vibration level at the current licensed thermal power will be compared to the acceptance criteria. Using this information, the vibration levels expected at the new higher power levels will be extrapolated. Confirm whether you will collect vibration data at 50% and 75% in addition to the 100% of the current thermal power level. If not, explain how do you extrapolate the vibration level at the new higher power level from only one data point at 100% power level.

Response to Question 11-16

Vibration data will be taken at approximately 50%, 75%, and at several power levels between 90 and 100% of CLTP. Data taken at these power levels should be sufficient to extrapolate expected vibration levels at 100% EPU rated thermal power conditions.

NRC Question 11-17

Discuss the functionality of safety related mechanical components (i.e., safety related valves and pumps, including air-operated valves (AOV) and safety and relief valves) affected by the proposed power uprate to ensure that the performance specifications and technical specification requirements (e.g., flow rate, close and open times) will be met for the proposed power uprate. Confirm whether and how safety-related AOVs and motor-operated valves (MOVs) will be capable of performing their intended function(s) following the proposed power uprate including such affected parameters as fluid flow, temperature, pressure and differential pressure, and ambient temperature conditions. Identify the mechanical components that were not evaluated at the uprated power level and the basis for your determination. Also, discuss the effect of the proposed power uprate on the pressure locking and thermal binding of safety related power-operated gate valves for Generic Letter (GL) 95-07. In reference to Section 4.1.5, provide a summary of your evaluation for the BSEP response to GL-96-06 for the proposed power uprate conditions.

Response to Question 11-17

As documented throughout the PUSAR (i.e., Enclosure 3 of CP&L's EPU amendment request (Serial: BSEP 01-0086, dated August 9, 2001)), on a system basis, the functionality of safety-related pumps and valves under EPU conditions was confirmed.

Since this EPU does not involve an increase in the operating pressure of the reactor, the affect on safety-related mechanical components is minimal. No changes in safety-related pump characteristics (i.e., flow or head) or assumed system response times are required. From a safety-related mechanical component perspective, the only change of significance from EPU is the increase in the post-LOCA containment pressures and temperatures as shown on Table 4-1 of the PUSAR.

The affects of these higher post-LOCA containment temperatures and pressures on the net positive suction head of safety-related pumps are discussed in Section 4.2.5 of the PUSAR and Enclosure 7 of CP&L's EPU amendment request.

There is no listing of the mechanical components that were not specifically evaluated or determined not to be impacted by EPU. However, PUSAR Section 6.8, "Systems Not Impacted by EPU," identifies those systems that were generically dispositioned as unaffected by EPU. For systems that are impacted by EPU, the components affected are discussed on a system by system basis throughout the PUSAR.

As part of an industry-wide initiative, BSEP is performing a systematic evaluation of AOVs, similar to that previously performed for MOVs under NRC Generic Letter 89-10, utilizing the Joint Owners' Group (JOG) evaluation methodology. That evaluation is independent of EPU, which was performed using the existing design basis of record. The current design basis for those safety-related AOVs that are subjected to containment pressure is based upon a differential pressure of 62 psig, which is significantly higher than the EPU peak accident pressure of 46.4 psig, see PUSAR Table 4-1. The MSIVs, which are subjected to increased steam flow, were evaluated as being bounded by the generic evaluation provided in Section 4.7 of ELTR2, see PUSAR Section 3.7. Therefore, it has been confirmed that safety-related AOVs are not adversely impacted by operation at EPU conditions.

As discussed above, there are no changes to the configuration of safety-related pumps that could result in higher differential pressures for MOVs. Additionally, since there is no change in the operating pressure of the reactor or an increase in SRV setpoints, see PUSAR Section 3.1, MOVs which are relied upon for isolation of high energy line breaks will be unaffected by EPU. The slight increase in post-LOCA containment pressures results in an increase in the maximum expected differential pressure for 56 safety-related MOVs. The EPU evaluation of these MOVs concluded that two MOVs (i.e., Unit 1 and 2 E11-F016A, RHR A-Loop Drywell Spray Injection Valves) will require torque switch adjustments, which will be completed prior to the implementation of EPU. Although no field adjustment is required for the other valves (i.e., the existing torque switch setting is adequate for the slightly higher differential pressure), the design calculations for all 56 affected valves will be revised to reflect the higher differential pressures. The MOV evaluation for EPU conditions also concluded that the impact on MOV terminal voltages and resultant stroke times is negligible.

In response to NRC Generic Letter 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," screening criteria were developed to identify any valves which could be susceptible to either pressure locking or thermal binding. Those screening criteria, which were previously approved by the NRC (i.e., Reference: NRC Letter dated

December 3, 1999, "Safety Evaluation of Response to Generic Letter 95-07, Brunswick Steam Electric Plant, Units 1 and 2 (TAC Nos. M93439 and M93440)"), are not being changed by EPU. No new valves were identified as being susceptible to pressure locking or thermal binding as a result of EPU. BSEP valves which were previously identified as being susceptible to pressure locking have had their discs drilled or bonnets vented to eliminate the potential. The slight increase in the post-LOCA containment temperatures as a result of EPU, slightly increases the differential temperature experienced by some valves. This slight increase in differential temperature is insignificant (i.e., less than 5°F over that assumed in the original Generic Letter 95-07 evaluation), and will not result in any new valves being susceptible to thermal binding based on the previously approved screening criteria. The implementation of EPU will not result in any new valves being susceptible to pressure locking or thermal binding as discussed in Generic Letter 95-07.

In response to Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," CP&L implemented plant modifications on the RHR shutdown cooling, drywell floor drain, and drywell equipment drain piping to eliminate the potential for post-accident over-pressurization of these closed sections of piping. The design solution for these three piping systems, a continuous vent to preclude over-pressurization, is not impacted by the slight increase in post-accident containment temperatures and pressures due to EPU. The remaining BSEP system to be addressed per Generic Letter 96-06 is the Reactor Building Closed Cooling Water (RBCCW) system (i.e., evaluate the potential water hammer concern during post-accident system restart). As discussed in Section 4.1.5 of the PUSAR, CP&L is participating in an industry collaborative project with the Electric Power Research Institute (EPRI) and the Nuclear Energy Institute (NEI) to develop a generic technical basis to address the water hammer issues. CP&L has previously committed to updating the NRC on the intended actions with respect to Generic Letter 96-06 after the NRC approves the EPRI/NEI generic technical basis. The final Generic Letter 96-06 evaluation for the RBCCW system will be performed based on EPU conditions.

NRC Question 11-18

In reference to Section 3.11, provide a summary addressing your evaluation of the effects of the proposed power uprate on the BOP piping, components, and pipe supports, nozzles, penetrations, guides, valves, pumps, heat exchangers and anchorages. Also, provide the calculated maximum stresses and fatigue usage factors for the most critical BOP piping systems, the allowable limits, the code of record and code edition used for the power uprate conditions. If different from the code of record, justify and reconcile the differences.

Response to Question 11-18

Evaluation methodology of pipe supports, nozzles, penetrations, guides, valves, pumps, heat exchangers and anchors at the EPU conditions is provided in Response to NRC Question 11-12.

The original piping design code for BSEP is USAS B31.1.0 - 1967. As noted in the following tables, some run of record pipe stress calculations were performed using ANSI B31.1 - 1973, which was previously reconciled with the original code of construction. No changes in the code, or code editions used in current run of record calculations, were made as a result of the EPU evaluations.

The USAS B31.1 Power Piping Code does not require the fatigue usage factor calculation, hence it is not evaluated for BOP piping.

Piping Systems potentially affected by EPU are:

1. Main Steam Outside Containment, Including Turbine Bypass and SRVDL Piping.
2. Residual Heat Removal Outside Containment.
3. Feedwater Outside Containment.
4. Core Spray Outside Containment.

Since the operating pressure of the reactor remains the same for EPU, the pressure and temperature conditions for the main steam piping does not change from the current design basis piping calculations. The EPU will result in an increase in steam flow of approximately 15% over current conditions. The consideration of steam flow transients on main steam piping, other than SRV discharge loads, is beyond the BSEP design basis. For BSEP, the consideration of these transient loads is specifically exempted by the Standard Review Plan (SRP), Section 10.3, "Main Steam Supply System," Revision 3, Subsection V.

Suppression pool temperature increases due to EPU for the long-term cooling application mode of RHR. However, the EPU temperature increase is enveloped by CLTP (i.e., analyzed) values. Pressure evaluations are based on design or peak pressure values. These pressure values are not changed or redefined as a result of EPU. There is no change in the system flow due to EPU. Therefore, the affects of EPU on RHR piping outside containment is bounded by the CLTP analysis.

The summaries of the feedwater and core spray piping evaluation results are provided in the following tables. No new computer codes were used or new assumptions were introduced for these evaluations.

Feedwater and Core Spray Piping Stress Summary - Piping Supports/Penetration Nozzles

Feedwater Piping Stresses

[Redacted - Four Tables]

Feedwater Supports, Nozzles, and Penetrations

Increase support loads due to EPU conditions are within the allowable For all the supports, nozzles, and penetrations associated with the feedwater piping.

Core Spray Piping Stresses

[Redacted - Thirteeen Tables]

Core Spray Supports, Nozzles, and Penetrations

Increase support loads due to EPU conditions are within the allowable For all the supports, nozzles, and penetrations associated with the core spray piping.

NRC Question 11-19

Referring to Sections 3.5 and 4.1.2, provide a discussion of the evaluation of piping systems attached to the torus shell, vent penetrations, pumps, and valves, that are affected by increased torus temperature and changes in LOCA dynamic loads (pool swell, condensation oscillation, and chugging) and increased temperature and flow in the main steam and feedwater systems due to the proposed power uprate. Identify supports and piping systems affected by required modifications as a result of the proposed extended power uprate.

Response to Question 11-19

The evaluation methodology, assumptions, and scaling factors, etc. for piping systems attached to the torus shell, vent penetrations, pumps, and valves are the same as those described in the Responses to NRC Questions 11-12, 11-13, and 11-18. Based on the detailed study, it was concluded that the effects of the increased torus temperature and changes in LOCA dynamic loads (i.e., pool swell, condensation oscillation, and chugging) on these piping systems, the design basis accident LOCA dynamic loads including pool swell loads, condensation oscillation loads, and chugging loads, at EPU conditions are bounded by CLTP loads.