ATTACHMENT 5

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To

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License Amendment Request 193

Component Evaluations for Steam Generators and Pressurizer (Non-Proprietary)

Kewaunee
Power Uprate
Engineering Report Sections 5.7 and 5.8

Non-Proprietary

5.7 Steam Generator Component Evaluations

Kewaunee nuclear power plant (KNPP) has proposed uprating their operating Nuclear Steam Supply System (NSSS) power level from 1,657.1 MWt (828.6 MWt/loop) to 1,780 MWt (890 MWt/loop). This represents a power uprating of 7.4 percent. To support the planned 7.4-percent power uprating, the existing pressurizer and Model 54F replacement steam generators (RSGs) have been evaluated for operation at the uprated-power conditions. The steam generator evaluations included steam generator tube plugging (SGTP) in the range from 0 to 10 percent, and considered both high- and low-average temperature (T_{avg}) conditions. Any steam generator or pressurizer design transients that were affected by the upgraded power levels were addressed in the evaluations

5.7.1 Steam Generators

The Kewaunee Model 54F RSGs were analyzed at the uprated-power conditions for thermal-hydraulic performance (including moisture carry over – MCO), structural integrity, tube vibration (including flow induced vibration), fatigue and wear, hardware changes and additions (repair hardware), loose parts, and tube integrity. It was concluded that the Model 54F steam generators will support operation at the uprated power levels. Detailed discussions regarding the evaluations and the conclusions reached for each aspect of steam generator operation are included in specific sections later in this report.

5.7.1.1 Thermal-Hydraulic Evaluation

This report section summarizes the thermal-hydraulic analyses performed to determine the operating characteristics of the KNPP Unit 1, Model 54F, RSGs to support the proposed 7.4-percent power uprate.

The analyses ensure that the thermal-hydraulic performance of the steam generators after the uprate remain in the acceptable range as compared to the current design operating conditions.

Based on thermal-hydraulic evaluations, the following observations are made:

1. Steam generators will be hydrodynamically stable; the damping factor is highly negative.

2. Moisture carryover remains below the design limit of 0.25% for all five cases analyzed.

3. The maximum calculated ratio of the local mixture quality to the predicted quality at DNB is 0.81. Thus for all analyzed conditions, the Kewaunee RSGs have sufficient DNB margin and therefore not expected to have local dry out on tube wall.

4. As presented in Table 5.7-1 all thermal-hydraulic parameters are within acceptable ranges for the 7.4% power uprate conditions with tube plugging level up to 10%.

5.7.1.1.1 Input Parameters and Assumptions

Steam Generator Geometry

All input variables other than the operating conditions needed for the GENF code are obtained from the thermal-hydraulic design data report for the Kewaunee Model 54F.

Operating Conditions

The reference case (referred to as Case 0) for this evaluation was established as the 100-percent power case (1,657.1 MWt NSSS power). All operating conditions, except the blow-down flow rate and water level, are obtained from Performance Capability Working Group (PCWG) parameters for the RSG Program. The blow-down flow rate of 36,000 lb/hr and normal water level of 493.6 inches above top of tube plate are taken from the RSG thermal-hydraulic report.

The 7.4-percent uprated-power conditions are defined by PCWG-2707. Cases 1 through 4 are evaluated for the 107.4-percent NSSS power of 1,780 MWt, and correspond to the four operating cases shown on PCWG-2707. Case 1 is with steam generator inlet temperature of 590.8°F and 0-percent tube plugging. Case 2 is with the same primary conditions and 10-percent tube plugging. Case 3 is with a higher steam generator inlet temperature of 606.8°F with 0-percent tube plugging, and Case 4 has the same primary operating parameters as Case 3 and 10-percent tube plugging. All cases assume the fouling factor of 110E-6 hr-ft^2-°F/BTU, blow-down flow rate of 36,000 lb/hr, and the normal water level. The GENF input data for all five operating conditions are summarized in Table 5.7-1. Note the vessel primary outlet or the hot leg (steam generator inlet) temperatures are used as input to the GENF code for thermal-hydraulic evaluations.

ATHOS Input Conditions

The model mesh and input files for the preprocessors ATHOGPP, PLATES, and ATHOS code (Reference 6) developed for a generic Model 54F steam generator analysis were used for this analysis. This generic model is representative of the Model 54 F generators at Kewaunee. The preparation of ATHOS input operating conditions for Case 2 is summarized below as an example.

Steam drum pressure = GENF calculated steam pressure + pressure drop across nozzle
= 637.84 psia + 5.054 psi = 642.894 psia = 4.43375 Mpa

Primary flow rate = 3.4682E7 lb/hr = 2189.457 kg/sec for half of steam generator

Primary coolant inlet temp = 590.8°F = 583.78 K

Feedwater flow rate = 3.9003E6 lb/hr = 246.23 kg/sec for half of steam generator

Feedwater inlet temperature = 437.1°F = 498.39 K

Blowdown flow rate = 36000 lb/hr = 2.273 kg/sec for half of steam generator

Water level from top of tube sheet = 493.6 in. =12.537 m

RRHOT = RRCOLD = CR/2 = 3.89/2 =1.995

The secondary side saturated vapor and liquid viscosity, conductivity, and specific heat values correspond to a pressure of 642.894 psia and are obtained from ASME steam tables (Reference 1).

Assumptions

Assumptions used in the GENF and ATHOS codes to predict the secondary side operating characteristics are assumed valid for this evaluation. The vessel outlet primary temperatures are used as input for the GENF thermal-hydraulic evaluations.

The KNPP has Model 54F steam generators with an improved Moisture Separation System. The separator improvements, field data, and methodology to predict separator performance are presented Reference 2. The methodology was assumed to be valid for operation at the uprated-power conditions.

5.7.1.1.2 Description of Analysis/Evaluation

The GENF code, Version 1.1.5, was used to calculate the secondary side thermal-hydraulic characteristics at both the 100-percent (Case 0) and 107.4-percent (Cases 1 through 4) power conditions. Moisture carryover (MCO) evaluations were performed using methodology discussed in Reference 2 for the KNPP Model 54F RSGs.

The GENF calculated operating conditions were utilized by the ATHOS code (Reference 6) to calculate the 3-D flow field parameters on the secondary side of the steam generator. The ATHOS analysis was utilized to determine the effects of the power uprating and tube plugging on the tube wall dry-out margin.

Method Discussion

Thermal-hydraulic conditions for Kewaunee steam generators are evaluated for the following five cases:

Case 0 – 100-percent power with 0-percent tube plugging, the current design basis from References 3 and 4.

Case 1 – 7.4-percent uprate plant parameters with the vessel average temperature, T_{avg}, of 556.3 °F (steam generator inlet temperature, T_{hot}=590.8°F) and 0-percent tube plugging from PCWG-2707 (Reference 5).

Case 2 – 7.4-percent uprate plant parameters with the vessel average temperature, T_{avg}, of 556.3 °F (T_{hot}=590.8°F) and 10-percent tube plugging from PCWG-2707.

Case 3 – 7.4-percent uprate plant parameters with the vessel average temperature, T_{avg}, of 573.0 °F (T_{hot}=606.8°F) and 0-percent tube plugging from PCWG-2707.

Case 4 – 7.4-percent uprate plant parameters with the vessel average temperature, T_{avg}, of 573.0 °F (T_{hot}=606.8°F) and 10-percent tube plugging from PCWG-2707.

The GENF code was used to calculate steady-state steam generator characteristics. The output from the GENF code includes various parameters such as primary temperatures, circulation ratio, steam flow rate, steam pressure, secondary side pressure drop, secondary fluid inventory, damping factor, etc. The GENF results are used to evaluate acceptability of steam generator performance with power uprate. The GENF results are also used in supplementary calculations discussed below.

Moisture Carryover Evaluation

Excessive MCO may result in erosion-corrosion problems in the steam piping and/or steam turbine. Prior MCO assessments (Reference 2) have been performed for the Kewaunee 54F RSGs. The KNPP RSGs comprise the original Model 51 steam generator upper shell with moisture separator and other improvements, and with a new lower part of the steam generator, including a new bundle with Inconel 690 tubes and 54,500 ft^2 heat transfer area. The separator improvements are described in Reference 2.

The RSG MCO assessment (Reference 2) also includes field data and correlations between separator parameter and MCO, as well as between the water level and MCO. The separator parameter is defined as: (steam flow rate, lbm/hr)^2 X (sp vol of steam at P_s,ft^3/lbm). The specific volume of vapor is a function of pressure and, therefore, accounts for variations in steam pressure. The MCO field data and correlations are included in Reference 2. The calculated values of MCO for Cases 0 through 4 are provided in Table 5.7-1.

Peak Heat Flux Evaluation

Peak heat flux on the hot-leg side of the tube bundle was calculated from GENF. The subtraction of fouling resistance in the following equation provided conservative results for heat flux

$$Q'' = (T_{hot} - T_s)/(R_t - R_f)$$

Q'' = Peak heat flux BTU/hr-ft^2
T_{hot} = Hot leg temperature - °F
T_s = Saturation temperature - °F
R_t = Total thermal resistance – hr-ft^2-°F/Btu
R_f = Fouling resistance – hr-ft^2-°F/Btu

Prediction of Secondary Side Mixture Quality at DNB

The ratio of the local secondary fluid mixture quality (X) to the quality at departure from nucleate boiling (X_{DNB}) in every flow cell of the ATHOS model is determined using the post-processor PLTATHOS. The correlations used to predict the quality at DNB are given in References 3 and 4. The maximum values of (X/X_{DNB}) for 100-percent power (Case 0) and 107.4-percent power (Case 2 and Case 4) are included in Table 5.7-1.

5.7.1.1.3 Acceptance Criteria

The relevant acceptance criteria for KNPP 7.4-percent power-uprate conditions are as follows:

- Secondary side operating characteristics remain within acceptable bounds at the power-uprate conditions (this is demonstrated for some parameters by showing that the change in the parameter is minor, as opposed to comparison to a fixed limit).
- There is no local dry-out on the tube wall.
- MCO remains below the design limit of 0.25 percent.
- The damping factor for hydro-dynamic instability evaluation is negative.
- Changes in primary and secondary fluid mass and heat content are small, 7% or less.
- The variations in the secondary fluid flow rates and velocities are also small, approximately 12% or less.
- Increases in the primary to secondary side heat fluxes are proportional to power uprate and tube plugging levels.

Results from the thermal-hydraulic analysis were utilized for U-bend tube wear and loose part evaluations that are discussed later in this report.

5.7.1.1.4 Results

The thermal-hydraulic evaluation of the KNPP Unit 1 Model 54F RSGs focused on the changes to secondary side operating characteristics at the 7.4-percent uprate conditions. The following evaluations were performed to confirm the acceptability of the steam generator secondary side parameters. The results of the evaluations are summarized in Table 5.7-1.

Bundle Mixture Flow Rate

The steam flow rate will increase with the 7.4-percent power uprate. With uprating, the GENF-calculated steam flow rate per generator increased from 3.55 to 3.88 million lb/hr, and the calculated circulation ratio decreased from 4.25 to 3.89. The secondary side flow rate in the tube bundle is the product of the circulation ratio and the steam flow rate. The resulting bundle flow rates are 15.10 and 15.04 million lb/hr respectively, or essentially the same at both 100 percent, and the four power uprate cases at 107.4-percent power.

The secondary fluid velocities in the U-bend region are 9 percent higher at the uprate conditions with the primary average temperature, T_{avg}, of 556.3°F, and 0-percent tube plugging. They are 6 percent lower at the uprate conditions with the T_{avg} of 573.0°F, and 0-percent tube plugging.

The 7.4-percent power uprate and the changes in T_{hot} and feedwater temperatures, T_{feed}, essentially have no effect on the secondary flow in the downcomer. The fluid velocities in the downcomer and at the wrapper opening are predicted to be within 1 percent of their values at 100-percent power.

Steam Pressure

The steam pressure is affected by the available heat transfer area in the tube bundle and the average primary fluid temperature With a 7.4-percent power uprate and the T_{avg} of 556.3°F, GENF calculated that the steam pressure would decrease from 669.4 psia to 660.0 psia. With the same T_{avg} and 10-percent tube plugging, the steam pressure decreased further to 637.8 psia. With 7.4-percent uprating and the T_{avg} of 573.0°F, GENF calculated steam pressures would be 777.4 psia with 0-percent tube plugging, and 752.9 psia with 10-percent tube plugging.

Heat Flux

Average heat flux in the steam generator is directly proportional to heat load, and inversely proportional to the heat transfer area in service. For the 0-percent tube plugging case, the calculated average heat flux increased from 51,902 BTU/hr-ft^2 at 100-percent power, to

55,721 BTU/hr-ft^2 at 107.4-percent power. With 10-percent tube plugging at the 107.4-percent uprated-power conditions, the average heat flux increased further to 61,912 BTU/hr-ft^2.

A measure of the margin for DNB transition in the bundle is a check of the ratio of the local quality to the estimated quality at DNB transition, or (X/X_{DNB}). The ATHOS analyses show that the maximum (X/X_{DNB}) increases from 0.74 at 100-percent power, to 0.81 at 107.4-percent power, with 10-percent tube plugging. The (X/X_{DNB}) ratio is less than 1.0, indicating sufficient margin from DNB, or local tube wall dry-out, and that the Kewaunee Model 54F RSG tube bundle operates in the nucleate boiling regime at 107.4-percent power-uprate conditions.

Moisture Carryover

Field tests for MCO, have been performed for Model 51 Moisture Separation System improvements (Reference 2). The KNPP Model 54F RSGs include the same Moisture Separation System improvements as the Model 51 units tested. Reference 2 includes field data and correlations between separator parameter and MCO, as well as between the water level and MCO. The correlations have been used to calculate conservative values of MCO for the Kewaunee Model 54F RSGs

The test results indicated that the separation system improvements are highly effective. The calculated carryover was 0.12 percent at full power, versus the design specification limit of 0.25 percent or less. The operating parameters, which can have an effect on moisture performance, are steam flow (power), vapor-specific volume (steam pressure), and water level. The MCO values for the Kewaunee power-uprate conditions were calculated from GENF results and the field data provided in Reference 2. The calculated MCO increased from 0.12 percent of steam flow at 100-percent power, to 0.19 percent at 107.4-percent power-uprate conditions, with 0-percent tube plugging. With 10-percent tube plugging, the maximum calculated MCO was 0.22 percent. All calculated MCO values are below the 0.25-percent limit at the 7.4-percent uprate condition.

Hydrodynamic Stability

The hydrodynamic stability of a steam generator is characterized by its damping factor. A negative value of the damping factor indicates that any disturbance to thermal-hydraulic parameters, such as flow rate or water level, will automatically reduce in amplitude, and the steam generator will return to stable operation. The damping factor decreases from numerically (actually an increase in damping) from –576.6 hr^{-1} at nominal power to a minimum value of –624.6 hr^{-1} when T_{hot} goes to 590.8 F. With in increase in T_{hot} to 606.8 F, the damping increases numerically to a maximum value of –547.9 hr^{-1} (a reduction in damping). This indicates that the Kewaunee RSGs will continue to operate in a hydro-dynamically stable manner when operating at the 107.4-percent uprated-power conditions.

Steam Generator Secondary Fluid Inventory

Secondary side fluid inventory consists of the mass of both liquid and vapor phases. The vapor mass is approximately 6 percent of total inventory. At the 7.4-percent uprated-power conditions, with 0-percent tube plugging, the calculated secondary fluid mass decreased from 98,168 lbs to 95,208 lbs, or approximately by 3 percent. The minimum calculated inventory of 94,357 lbs is for a T_{avg} of 556.3°F with 10-percent tube plugging at 107.4-percent power. The small changes in inventory are judged to have no effect on operation.

Steam Generator Secondary Side Pressure Drop

The calculated secondary side pressure drop increased from 18.0 psi to 21.3 psi as a result of a 7.4-percent power uprate. It further increased to 21.6 psi at a T_{avg} of 556.3°F with 10-percent tube plugging at 107.4-percent power. With the higher T_{avg} of 573.0°F at 107.4-percent power the pressure drop was calculated to be 19.8 psi with 0-percent tube plugging, and 20.1 psi with 10-percent tube plugging. The small increase in pressure drop would have no significant effect on the feed system operation.

5.7.1.1.5 Conclusions

Based on the thermal-hydraulic evaluations for operation at the 7.4-percent uprated-power conditions, the following conclusions were drawn:

- The steam generators remain hydro-dynamically stable; the damping factor is highly negative, varying from –547.9 hr^{-1} to –624.6 hr^{-1} for the five cases analyzed.
- MCO remains below the design limit of 0.25 percent for all five cases analyzed.
- The Kewaunee RSGs have sufficient DNB margin for all analyzed conditions and, therefore, are not expected to experience local dry-out on any tube wall.

In conclusion, all calculated thermal-hydraulic parameters of the Kewaunee Unit 1 RSGs will remain within acceptable ranges for operation at the 7.4-percent uprated-power conditions with tube plugging levels of up to 10 percent. The thermal-hydraulic characteristics of the Kewaunee Model 54F RSGs at the 7.4-percent uprated-power conditions are summarized in Table 5.7-1.

5.7.2 Structural Integrity Evaluation

Evaluations have been performed to consider the effects of a 7.4-percent Power Uprating on the structural integrity of the RSGs at the KNPP. The Kewaunee RSG power uprate evaluation was based on the existing analyses and evaluations from the previous SGR Project. The PCWG operating parameters for the Kewaunee RSG 7.4-percent uprate conditions are summarized in PCWG-2707. It has been determined that the existing RSG structural analyses for all components affected only by the primary side or secondary side steam temperatures (T_{steam}) and pressure differentials remained applicable for the uprated conditions, and did not require any revision to support the plant uprating.

Only components that are affected by the revised feedwater temperatures (T_{feed}) and flow rates associated with the 7.4-percent uprate required further evaluation. The affected analyses include the feedwater nozzle and thermal sleeve analysis (Reference 5) and the J-nozzle-to-feedring weld fatigue analysis (Reference 6). The feedring seismic and steam line break analysis (Reference 7) is not affected by the T_{feed} changes associated with the uprate. Per Reference 7, the most significant response with respect to transient thermal stresses in the feedwater system occurs in the nozzle and thermal sleeve region. Thermal and cyclic stresses in the feedring and fittings, beyond the nozzle region, are bounded by the structural evaluation of the feedwater nozzle and thermal sleeve in Reference 5. The following discussion addresses the effects of the revised feedwater transients associated with the plant uprate on the stresses and fatigue usages calculated in References 5 and 6. The structural integrity of all other steam generator primary and secondary side components continue to be demonstrated by the existing analyses performed in support of SGR (References 8 through 10).

5.7.2.1 Input Parameters and Assumptions

The PCWG parameters applicable to the Kewaunee 7.4-percent Uprate Project have been defined in PCWG-2707. Based on the Uprate Program's intent to utilize existing analyses from the RSG Program, a review of the applicability of the existing RSG analyses has been performed in consideration of the NSSS design parameters. Based on the philosophy of selecting operating conditions for the uprate that will be bounded by the existing and previously analyzed RSG conditions, the analyses in support of the Kewaunee Uprate Project are limited to those impacted by the increase of T_{feed}. For all transients seeing primary side temperatures T_{hot} and T_{cold}, the transient histories developed for the RSG Program bracket those for the uprating. (This is because the RSG Program has a larger temperature window for design purposes and the limiting full-power T_{hot} and T_{cold} values for the uprated conditions are bounded by the ones used in the RSG analyses).

Similarly, for all transients seeing the RCS pressure, the transient histories remain valid for the uprated conditions. For all transients seeing T_{steam}, with a lower limit of 644 psia is to be placed on the minimum steam pressure, the RSG transient histories for the RSG Program also bracket those for the uprating.

On this basis, the majority of the structural analyses performed in support of the Kewaunee RSG Project remain applicable for the uprated condition. Exceptions include the analyses of the feedwater nozzle and thermal sleeve (Reference 5), and the feedring-to-J-nozzle weld fatigue analysis (Reference 6).

5.7.2.2 Description of Analyses/Evaluations

The analyses of the critical components for the primary and secondary sides of the Kewaunee Model 54F RSGs were previously performed for operating conditions defined by the design specification for the RSG and have been documented in References 7, 8, 9 and 10. Of these analyses, only Volumes 1 and 2 of Reference 8 have been determined to be impacted and require further evaluation as a result of the operating conditions associated with the 7.4-percent plant uprate. These volumes address the structural integrity of the feedwater nozzle and thermal sleeve (Reference 5) and the fatigue analysis of the J-nozzle-to-feedring weld (Reference 6). For these reports, evaluations have been performed to determine the impact of changes in T_{feed} at steady state conditions and for certain transients.

The minimum T_{feed} for the uprated conditions are unchanged from the original RSG parameters (for all transients), while the maximum T_{feed} for the uprated conditions has increased for certain transients. The durations of all transients have remained the same, as do the times within the transients where step changes or ramps in T_{feed} and flow occur.

For the unit loading and unloading, loss of load, loss of power, loss of flow, and reactor trip events, the minimum T_{feed} remains at 32°F, while the maximum T_{feed} (occurring at either the beginning or end of the transient) increases from 427.3°F to 437.1°F. This is the case for both the low-temperature and high-temperature conditions. For the large step-load decrease event, the minimum T_{feed} remains unchanged at 200°F for the uprated conditions. The maximum T_{feed} (which occurs at the beginning of the large step-load decrease transient) increases from 427.3°F to 437.1°F. Again, this applies for both the low-temperature and high-temperature conditions. The net increase in the overall range of T_{feed} for this event is also 9.8°F.

For the analysis of the feedwater nozzle and thermal sleeve, in order to demonstrate the acceptability of the maximum ranges of stress intensity ranges at critical analysis sections (referred to by Analysis Section Number – ASN) for the uprated conditions, scale factors were developed to account for the slight increase in the range of T_{feed} for the events listed above, as well as to account for small effects this change could have on temperature-dependent film coefficients that were developed in Reference 5 for the feedwater nozzle and thermal sleeve.

5.7.2.3 Acceptance Criteria

The acceptance criteria for the the feedwater nozzle and thermal sleeve and the J-nozzle-to-feedring weld are that the maximum range of stress intensity and cumulative fatigue usage factor at each analysis section (See Figures 5.7-1 and 5.7-2 for a definition of ASNs) satisfy the ASME Code (Reference 25) specified allowables. In addition, the cummulative fatigue usage factor must be less than or equal to unity.

For certain ASNs in the original feedwater nozzle and thermal sleeve analysis (Reference 5), the maximum range of stress intensity calculated in Reference 5 already exceeded the Code allowable of $3S_m$. For those cases, the appropriate K_e factors were developed in Reference 5, in accordance with Subsection NB-3228.5 of the ASME Code for use in the fatigue analysis. The evaluation for uprated conditions assessed whether the changes in T_{feed} resulted in any further increase in the maximum range of stress intensity at these critical ASNs, and if so, new K_e factors were applied to determine the revised fatigue usage factors at those locations.

5.7.2.4 Results

As described above, scale factors were developed to account for the slight increase in the range of T_{feed} for the events listed above, as well as to account for small effects this change could have on temperature-dependent film coefficients that were developed for the RSG evaluations (Reference 5) performed for the feedwater nozzle and thermal sleeve.

For the uprated conditions, the maximum stress intensity ranges from the RSG feedwater nozzle and thermal sleeve evaluations (Reference 5) were conservatively multiplied by a scale factor of 1.04 for the stress intensity ranges involving unit loading or unloading, loss of load, loss of power, and reactor trip. A scale factor of 1.06 was applied for the large step-load decrease event. The resulting maximum ranges of stress intensity were compared to the previously calculated values from Reference 17, and to the ASME Code allowable ($3S_m$) in Tables 5.7-2 and 5.7-3. As was done for the RSG evaluations for certain critical locations where the stress intensity ranges exceed $3S_m$, the stress ranges are determined to be acceptable once thermal-bending stresses are removed, per the simplified elastic-plastic analysis criteria of Subsection NB-3228.5 of the ASME Code. Revised K_e factors were calculated in accordance with the guidelines of the Code, and were applied to determine the effect on fatigue usage factors at these locations for the uprated conditions. On this basis, all of the maximum stress intensity ranges are acceptable with respect to the limits of the ASME Code (Reference 11).

Of the fifteen limiting stress locations analyzed in the original RSG feedwater nozzle and thermal sleeve analysis, shown in Figure 5.7-1, thirteen of these locations had low cumulative fatigue usage factors (CUF's) ranging from 0.03 to 0.42. From Table 1-1 of Reference 5, the most limiting fatigue usage factors were calculated at ASN 11 (CUF = 0.77) and ASN 15 (CUF = 0.86). For the uprated conditions, the revised fatigue usage factors at these locations showed only a small change associated with the slightly higher maximum feedwater temperature; the revised cumulative fatigue usage factors for the uprated conditions are 0.84 at ASN 11 and 0.94 for ASN 15. Since the other thirteen ASNs evaluated in Reference 5 had significant margin relative to the ASME Code limit of 1.0, revised CUF's were not calculated for these locations, because the relatively small increases in stress intensity due to the small increase in feedwater temperature for certain transients would still result in values well below the Code allowable of 1.0. Therefore, it is concluded that for all of the ASNs evaluated in Reference 5, the cumulative fatigue usage factors for the uprated conditions satisfy the cumulative fatigue usage requirements of the ASME Code.

For the J-nozzle-to-feedring weld, review of the structural analysis in Reference 6 indicates that the maximum stress intensity ranges for all of the limiting transients result at times when the feedwater has reached, or is near, its minimum value, which is unchanged for the uprated conditions. The thermal stresses in the feedring, J-nozzle, and attachment weld are primarily caused by the temperature difference between the secondary side (T_{steam}) outside the feedring, and the cold feedwater inside these components. On this basis, it was concluded that the T_{feed} changes for the uprated conditions (which affect only the maximum T_{feed} at times far removed from when the maximum stress intensity ranges occur) have no effect on the maximum ranges of stress intensity at the locations previously analyzed in Reference 18. The maximum stress intensity ranges at the limiting locations, shown in Figure 5.7-2, are summarized in Table 5.7-4.

For the fatigue analysis of the J-nozzle-to-feedring weld, it was concluded that the only T_{feed} change that could impact a portion of the fatigue analysis for the J-nozzles would be the slightly higher temperature associated with the steady-state temperature at time 0 for the large step-load decrease event. The stress intensity at the inside surface of the weld (ASN 3 in Figure 5.7-2, which produced the most limiting fatigue usage factor) decreased slightly, leading to a small increase in the stress intensity range for load combinations involving this event.

The fatigue calculations of Reference 6 were revised accordingly for affected load combinations. It was found that the fatigue usage factor at the limiting location—the inside surface of the weld (ASN 3 in Figure 5.7-2) increased only slightly, from a value of 0.528 (from Reference 6) to a value of 0.537 for the uprated conditions. This change is insignificant with respect to the Code allowable of 1.0.

5.7.2.5 Conclusions

Only those components impacted by the changes in T_{feed} associated with plant uprating were re-analyzed to demonstrate the acceptability of small changes in stress-intensity ranges and cumulative-fatigue usage factors associated with plant uprating. All other components that experience only the primary or secondary side temperature and pressure gradients are still bounded by the structural analyses performed for the RSGs when operating at the uprated-power conditions. Revised stress-intensity ranges and fatigue-usage factors for the limiting locations in the feedwater nozzle and thermal sleeve, as well as the J-nozzle-to-feedring weld, have been shown to remain less than the ASME Code allowable limits for the 7.4-percent uprated conditions. Therefore, the steam generators are shown to satisfy the requirements of the ASME Code and will maintain their stuctural integrity for the uprated conditions.

5.7.3 Primary-to-Secondary Pressure Differential Evaluation

This analysis evaluates the structural acceptability of primary-to-secondary side pressure differentials (ΔPs) for the Kewaunee Model 54F RSGs for transient conditions applicable to a 7.4-percent Power Uprating. The Kewaunee Model 54F steam generators are designed for a peak pressure differential value of 1,800 psi. As a result of the 7.4-percent Power Uprating, certain normal and upset operating condition transients may result in the design ΔP limit being exceeded.

The purpose of this analysis was two-fold:

- To determine if the ASME Code limits on design primary-to-secondary pressure drop are exceeded for any of the applicable transient conditions.
- If the limits on the design primary-to-secondary pressure drop are exceeded, to determine the minimum acceptable full-power steam pressure so that the pressure limits are satisfied.

5.7.3.1 Input Parameters and Assumptions

The full-power normal operating plant parameters applicable to the Kewaunee 7.4-percent Uprate Project are defined in PCWG-2707, with the exception of T_{steam} for the low T_{avg} operating condition. T_{steam} was limited to the value that was previously shown to be acceptable at the time of the RSG qualification. Limiting T_{steam} to the previously approved value permitted the analysts to utilize analyses performed as part of the design qualification for the RSG. For RSG design qualification analyses, the operating parameters were defined in Revision 6 of the Kewaunee Steam Generator Design Specification.

Similarly, the transient parameters applicable to the Uprate Project are defined as the 7.4-percent uprated power transient set. For situations where the RSG design qualification analyses did not bound the uprated power parameters, the transient parameters defined in Design Specification 414A03 were to be used.

In calculating the primary-to-secondary pressure drops for the small step-load increase and the small step-load decrease transients, the following conservative assumptions were used:

- The 10-percent small step-load increase transient may be initiated at any power level between 15-percent and 90-percent of full power. For this analysis, it was conservatively assumed that the transient was initiated from the 90-percent power level. This resulted in the highest primary-to-secondary pressure drop for this transient.

- The 10-percent small step-load decrease transient may be initiated at any power level between 100-percent and 25-percent of full power. For this analysis, it was conservatively assumed that the transient was initiated from the 100-percent full-power level. This resulted in the highest primary-to-secondary pressure drop for this transient.

5.7.3.2 Description of Analysis/Evaluation

The normal full-power operating parameters applicable to this analysis are summarized in Table 5.7-5. Conditions are defined for both high-temperature operating conditions (high T_{ave}) and low-temperature operating conditions (low T_{ave}). Note that operating parameters are also defined for plugging levels of 0 percent and 10 percent. However, because the 10-percent plugging conditions resulted in higher primary-to-secondary pressure gradients, only the 10-percent plugging level was evaluated.

5.7.3.3 Acceptance Criteria

The design pressure limit for primary-to-secondary pressure differential is 1,800 psi, as defined in the applicable design specification. The design pressure requirements for Class 1 equipment are defined in the applicable edition of the ASME B & PV Code Section III (Ref. 11) for the Kewaunee steam generators. The normal/upset operating transient conditions are subject to the following design pressure requirements.

- Normal Condition Transients: Primary-to-secondary pressure gradient will be less than the design limit of 1,800 psi.

- Upset Condition Transients: If the pressure during an upset-operating condition transient exceeds the design pressure limit, the stress limits corresponding to design conditions apply using an allowable stress intensity value of 110 percent of those defined for design conditions. In other words, as long as the upset-operating condition transient pressures are less than 110 percent of the design pressure values, no additional analysis is necessary. For the Kewaunee steam generators, 110 percent of the design pressure limit corresponds to 1,980 psi.

5.7.3.4 Results

The maximum primary-to-secondary differential pressures for high T_{ave} conditions were found to be 1,552 psi and 1,511 psi for normal and upset conditions, respectively (Ref. 27). For low T_{ave} conditions, the maximum pressure differentials are 1,630 psi, and 1,605 psi for normal and upset conditions, respectively. These values were all below the applicable design pressure limits of 1,800 psi for normal conditions, and 1,980 psi for upset conditions.

5.7.3.5 Conclusion

Based on the above analysis results, it is concluded that the design pressure requirements of the ASME Code are satisfied for the 7.4-percent uprate.

5.7.4 Tube Vibration and Wear

The impact of the proposed 7.4-percent uprate on the steam generator tubes was evaluated based on the current design basis analysis, and included the changes in the thermal-hydraulic characteristics of the secondary side of the steam generator resulting from the uprate. The effects of these changes on the fluid-elastic instability ratio and amplitudes of tube vibration due to turbulences have been addressed. In addition, the effects of the uprate on potential future tube wear have also been considered.

5.7.4.1 Input Parameters and Assumptions

The baseline tube vibration and wear analysis results for the Kewaunee Model 54F RSG are reported in Reference 12. The original vibration analysis demonstrated that the maximum fluid-elastic stability ratio for the expected tube support conditions was less than the allowable limit of 1.0. The original tube vibration analysis also determined that negligible tube responses occurred due to the vortex-shedding mechanism. The amplitudes of vibration due to turbulence were found to be reasonably small, with maximum displacements found to be on the order of a few mils (8 mils for the most limiting condition). The maximum expected tube wear that could occur over the remaining period of operation was found to range from ~3 to 6 mils, depending upon actual fit up, length of operation, and actual operating conditions.

5.7.4.2 Description of Analyses and Evaluations

The results of the current design basis vibration and wear analysis were modified to account for anticipated changes in secondary side thermal-hydraulic operating conditions due to the uprated-power conditions. Previously established values of fluid-elastic instability, turbulent amplitudes of vibration, and tube wear were modifed to incorporate the new operating parameters.

5.7.4.3 Acceptance Criteria

The acceptance criteria consists of demonstrating that the wear rate will not result in premature failure of a significant number of tubes during the steam generator operating life.

5.7.4.4 Results

For the expected support conditions, it was found that straight leg stability ratios were not significantly impacted. However the stability ratios for U-bend conditions increased from approximately 0.73 to 0.77, which is still less than the allowable limit of 1.0. As a result, the analysis indicated that large amplitudes of vibration are not projected to occur due to the fluid-elastic mechanism while operating the steam generator in the uprated-operating condition.

The maximum displacement values for turbulence excitation calculated in the original analysis were modified to account for uprate-induced changes in the operating conditions. For the most limiting tube-support condition, it was determined that the turbulence-induced displacement could increase from ~8 mils to ~11 mils. Displacements of this magnitude are not sufficient to produce tube-to-tube contact. However, the potential for tube wear must be considered.

As in the original analysis, the vortex shedding mechanism was found not to be a significant contributor to tube vibration. The potential for tube wear was addressed in the original analysis, and addressed wear in both the straight leg and U-bend portions of the steam generator. These calculations were then updated to reflect operation of the steam generators in an uprated-power condition The uprated-power calculation determined that the level of tube wear that could occur would increase from ~3 mils to ~4 mils at the uprated conditions. From these calculations, it can be concluded that although there may be an increase in the level of wear that would occur at the uprated-operating conditions, the increased level would not be significant. Any increase in the rate of tube wear would progress over many cycles and would be observable during normal eddy current inspections.

5.7.4.5 Conclusions

The analysis of the Kewaunee Model 54F RSGs indicates that significant levels of tube vibration will not occur from either the fluid-elastic, vortex-shedding, or turbulent mechanisms as a result of the proposed uprate. In addition, the projected level of tube wear as a result of vibration would be expected to remain small and will not result in unacceptable wear.

The analysis of the effects of the uprated power condition on the S/G tubes has addressed both the straight leg and U-bend portions of the S/G. Limiting wear calculations have been performed where it has been determined that the maximum projected rate of tube wear would increase no more than 25% over the current levels of wear. Since significant tube wear is not currently occurring at Kewaunee, a 25% increase will not be significant. Should significant tube wear initiate sometime in the future, the rate of tube wear would be sufficiently small such that any tubes requiring repair would be detected during the normal eddy current inspection program.

With respect to the effect of increased primary to secondary side pressure difference, it should be noted that there is no direct correlation of flow-induced vibration with primary to secondary side pressure differences. The S/G tubes respond primarily to the conditions associated with the secondary side since the forcing functions associated with the secondary side of the S/G dominate over any other effects. Any effects of primary-to-secondary side pressure difference are inherently considered in the analysis in that the secondary side conditions are defined by the total S/G conditions such as steam pressure, flow rates, re-circulation, etc., and includes the primary-to-secondary side pressure difference.

Note that in some model steam generators particular consideration is given to the potential for high cycle fatigue of U-bend tubes. This phenomenon has been observed in tubes with carbon steel support plates where denting or a fixed tube support condition has been observed in the upper most plate. However, since the Kewaunee S/G tube support plates are manufactured from stainless steel, there is no potential for the necessary boundary conditions (i.e. denting) to occur at the uppermost support plate. Hence, high cycle fatigue of U-bend tubes will not be an issue at Kewaunee.

5.7.5 Evaluations for Repair Hardware

The Kewaunee RSGs entered service in the Fall of 2001. There were no shop-welded plugs installed in either of the two steam generators. However, in anticipation of a possible future need to install a field-weld plug, an analysis was performed to qualify a Westinghouse field-installed weld plug. Also for possible future needs, both long and short 7/8-in. Westinghouse ribbed mechanical plugs were qualified for installation in the Model 54F RSGs at Kewaunee for the 7.4-percent uprated-power-operating conditions. In addition, since there are circumstances that may require tube ends to be reamed, a 0.020-in. wall thickness tube undercut (40-percent wall reduction) is considered and the reduced weld joint geometry qualified. Note that the evaluations in this section only address tube repair hardware provided by Westinghouse. Hardware provided by other vendors must be qualified by the vendor for use at the uprated power operating conditions.

5.7.6 Mechanical Plugs

The enveloping condition for the Westinghouse mechanical plug (Alloy 690 shell material) is the one that results in the largest pressure differential between the primary and the secondary sides of the steam generator. Both the PCWG parameter changes and the NSSS design transients were used to determine the effect of the power uprating on the mechanical plugs. The most critical set of parameters for the mechanical plug evaluation was determined to be the primary side hydrostatic pressure test in which the differential pressure across the plug is 3,107 psi (which remains unchanged after the power uprating).

5.7.6.1 Input Parameters and Assumptions

The mechanical plug evaluated for Kewaunee is the Westinghouse ribbed mechanical plug, 7/8-in. diameter, long and short lengths The plug material is SB-166, Alloy 690, as described in ASME Code Case N-474-1.

The tube material for the Kewaunee RSG tubes is SB-163, Alloy 690, as defined in ASME Code Case N-20-3.

5.7.6.2 Description of Analysis/Evaluation

A structural evaluation was performed for the Westinghouse mechanical plug for the 7.4-percent uprate condition. This evaluation was performed in accordance with the applicable requirements of the ASME B&PV Code (Reference 11). The first part of the evaluation dealt with stress and plug retention, and the second part of the evaluation addressed fatigue exemption condition compliance.

Structural evaluations for mechanical plug installations have been performed for installations at various plants. The approach for Kewaunee was to utilize a generic calculation that qualified the mechanical plug, and adjust the stress results to account for any primary-to-secondary pressure increase and/or tube-sheet geometry differences.

The critical parameter from the design of the plugs is the primary-to-secondary differential pressure. The plug shell was qualified for a 2,485 psi ΔP design condition. This design ΔP bounds all maximum normal and upset differential pressures calculated for the 7.4-percent uprating. The primary-to-secondary differential is actually limited to 1,800 psi for normal design conditions. The 1,800 psi value is well below the design pressure of 2,485 psi.

Since a mechanical plug is a part that is installed into the steam generator after entering service and is not part of the original steam generator, this part is typically fabricated to the requirements of the 1989 ASME Code Edition (Ref. 15). On this basis, the evaluation was conducted based on the 1989 Code year requirements. It was determined that the mechanical plug is also acceptable for the 7.4-percent uprate, based on the 1989 ASME Code Edition.

5.7.6.3 Acceptance Criteria

The Westinghouse mechanical tube plug was evaluated for the applicable transients associated with plant uprating. The primary stresses due to design, normal, upset, faulted, and test conditions must remain within the respective ASME Code-allowable values (Reference 11).

The cumulative fatigue must be less than, or equal to unity, or the ASME fatigue exemption rules must apply for a 40-year fatigue life for the plug. In addition to the stress criteria, plug retention must be ensured.

5.7.6.4 Results

The mechanical plug was evaluated for a maximum primary-to-secondary differential pressure of 2,485 psi. All stress/allowable ratios are found to be less than unity, indicating that all primary stress limits are satisfied for the plug shell wall between the top land and the plug end cap. The plug design was shown to meet the Class 1 fatigue exemption requirements per NB-3222.4 of the ASME Code (Reference 11). It was also determined that adequate friction is available to prevent the plug from dislodging for the limiting steady-state and transient loads.

5.7.6.5 Conclusions

Results of the analyses performed for the mechanical plug for Kewaunee show that both the long and short mechanical plug designs satisfy all applicable stress and retention acceptance criteria at the 7.4-percent power-uprate condition.

5.7.7 Weld Plugs

There are no shop-weld plugs installed in either of the two Kewaunee Model 54F RSGs at this time. However, there is a possibility that during the life of the steam generators there will be a need to install field-weld plugs. The Westinghouse field-weld plug designated for installation in the Model 54F steam generators is the model NPT-80. The structural evaluation of the weld plug addressed the qualification of the mechanical plug, based on applicable design transients for the 7.4-percent uprated-power conditions.

Note that this current evaluation only addresses the structural analysis and qualification of the weld plug. Prior to installation, the weld process for plug installation is to be developed and qualified The weld process development and qualification activity were not included as part of this 7.4-percent uprated-power evaluation.

5.7.7.1 Input Parameters and Assumptions

The Westinghouse NPT-80 weld plugs are fabricated from ASME SB-166, Alloy 690 rod material, as described in ASME Code Case N-474-1 (Ref. 13). The minimum yield for this material is 35,000 psi.

The tube material for the steam generator tubes is SB-163, Alloy 690, as defined by ASME Code Case N-20-3 (Ref. 14).

5.7.7.2 Description of Analysis/Evaluation

A structural evaluation was performed for the weld tube plugs for the 7.4-percent uprated-power conditions. The evaluation was performed to the applicable requirements of ASME B&PV Code (Reference 11).

The design condition was evaluated first. A vertical failure plane around the perimeter section of the weld plug was considered. The maximum design condition primary-to-secondary pressure differential of 1,800 psi was evaluated. The maximum secondary-to-primary pressure of 670 psi was also considered. The evaluation determined that the maximum secondary-to-primary pressure of 670 psi actually controls for the weld qualification. Stresses were found acceptable for design conditions.

Test conditions for the primary hydrostatic and secondary hydrostatic tests were then evaluated. Note that these test conditions were not affected by the uprated power. Values for primary stresses, primary-plus-secondary stresses, and primary-to-secondary stress range intensities were calculated. All stress values were found to be acceptable.

The normal/upset conditions were also reviewed. The generic evaluation of the NPT-80 weld plug determined that the controlling transient for both the normal and upset conditions was the loss-of-power transient. The differential pressure considered was 1,700 psi. This was the controlling pressure condition for the baseline transient conditions. The governing differential pressure for the 7.4-percent uprate for normal/upset conditions was calculated at 1,630 psi, which is bounded by the baseline 1,700 psi differential pressure. It was shown that the stress limits are acceptable for a 1,700-psi differential pressure.

The last step in the evaluation process was consideration of fatigue. An existing generic analysis for the NPT-80 weld plug includes a fatigue analysis calculation. For the uprated-power evaluation, the approach was to apply scaling factors to the existing analysis results to determine a revised fatigue usage factor.

5.7.7.3 Acceptance Criteria

Westinghouse shop-weld plugs were evaluated for the effects of changes to the plant design transients that occur due to the uprating. The primary stresses due to design, normal, upset, faulted, and test conditions must remain within the respective ASME Code allowable values (Reference 11). The maximum normal and upset primary-plus-secondary stress intensities are to be less than the $3S_m$ limit. The cumulative fatigue usage must be less than, or equal to unity.

5.7.7.4 Results

The evaluations for the uprated-power condition determined that the calculated stresses remain within the ASME Code limits for the design, normal/upset, and test conditions. A fatigue usage factor of 0.0021 was calculated, which is well below the allowable of 1.0. Therefore, it was concluded that the welded plug does meet the ASME Code cycle load fatigue limits for the 7.4-percent uprated-power conditions.

5.7.7.5 Conclusions

All primary stresses are satisfied for the weld between the weld plug and the tube-sheet cladding. The primary-plus-secondary stresses are found acceptable. The maximum primary-plus-secondary stress intensity was found to be acceptable. The cumulative fatigue usage factor was found to be less than 1.0. On this basis, the weld plug was found to be acceptable for use in the Kewaunee RSGs at the 7.4-percent uprated-power conditions.

5.7.8 Tube Undercut Qualification

The field machining of steam generator tube ends may by required to facilitate making modifications and repairs to tubes (that is, plugging, sleeving, and tube end reopening). Removal of the Westinghouse mechanical plug could potentially require a portion of the tube and weld material to be removed. This would be accomplished by a machining process (drilling and reaming). This evaluation addressed the acceptability of a 0.020 in. of tube wall thickness undercut (40-percent wall reduction) for operation at the 7.4-percent uprated-power condition.

5.7.8.1 Input Parameters and Assumptions

The tube material for the steam generator tubes is SB-163, Alloy 690, as defined by ASME Code Case N-20-3 (Ref. 14).

5.7.8.2 Description Of Analysis/Evaluation

A structural evaluation was performed for the undercut of the steam generator tube ends for the 7.4-percent uprate condition. The evaluation was performed to the applicable requirements of ASME B&PV Code (Reference 11). Past structural evaluations for steam generator tube end machining have been performed. The approach for the Kewaunee tube end evaluation was to utilize the results from an existing evaluation and adjust the existing stress values as appropriate for applicable design transients for the uprate. The adjustment value was conservatively based on the maximum increase in differential pressure across the tube sheet for 7.4-percent uprated-power operation.

A similar approach of applying factors as that taken for the calculation of stress values was utilized in the investigation of fatigue for the tube undercut machining.

5.7.8.3 Acceptance Criteria

The steam generator tube end undercut must be evaluated for the effects of the design transients that are applicable for the uprated-power conditions. The primary stresses due to design loading must remain within the respective ASME Code allowable values (Reference 25). The maximum range of stress intensities are to be less than the ASME Code $3S_m$ limit. The cumulative fatigue usage must be less than, or equal to unity.

5.7.8.4 Results

The results obtained found that all revised stresses for the 7.4-percent uprate condition are all within ASME Code allowable values. The maximum range for stress intensity was 79.42 ksi which occurred for the tube leak test/loss of load transient event combination. This compares to an ASME Code allowable of 79.80 ksi. It was found that cumulative fatigue usage values, when adjusted for the 7.4-percent uprate, remain acceptable. The maximum cumulative fatigue usage factor was calculated as 0.228, which remains less than the allowable factor of unity.

5.7.8.5 Conclusions

The 7.4-percent uprated-power stress evaluation of tube undercut in the Kewaunee Model 54F steam generators determined that the stresses are all within ASME Code (Ref. 11) allowable values. The fatigue usage values were found to be less than 1.0, therefore a 0.020-in. tube wall thickness undercut is acceptable for operation at the 7.4-percent uprated-power conditions.

5.7.9 Generic Evaluation of Loose Parts

The Kewaunee Model 54F RSGs are at an early stage in their service life. No loose parts are currently present in their generators at this time. A generic loose parts evaluation has been prepared addressing undefined loose parts in the generator, operating at the 7.4-percent uprated-power conditions Results of the Loose Parts evaluation are presented in Westinghouse WCAP-15941 (Ref 16).

5.7.10 Tube Repair Limits (Regulatory Guide 1.121 Analysis)

The heat transfer area of steam generators in a (PWR) NSSS comprises over 50 percent of the total primary system pressure boundary. The steam generator tubing, therefore, represents a primary barrier against the release of radioactivity to the environment. For this reason, conservative design criteria have been established for the maintenance of tube structural integrity under the postulated design-basis accident condition loadings in accordance with Section III of the ASME Code

Over a period of time, under the influence of the operating loads and environment in the steam generator, some tubes may become degraded in local areas. Partially degraded tubes are satisfactory for continued service provided defined stress and leakage limits are satisfied, and the prescribed structural limit is adjusted to take into account possible uncertainties in the eddy current inspection, and an operational allowance is made for continued tube degradation until the next scheduled inspection.

The NRC Regulatory Guide 1.121 (Reference 17) describes an acceptable method for establishing the limiting safe condition of degradation in the tubes beyond which tubes found defective by the established in-service inspection should be removed from service. The level of acceptable degradation is referred to as the "repair limit."

An analysis is being performed to define the structural limits for an assumed uniform thinning mode of degradation in both the axial and circumferential directions. The assumption of uniform thinning is generally regarded to result in a conservative structural limit for all flaw types occurring in the field. The allowable tube repair limit, in accordance with Regulatory Guide 1.121, is obtained by incorporating into the structural limit a growth allowance for continued operation until the next scheduled inspection, as well as an allowance for eddy current measurement uncertainty. Calculations have been performed to establish the structural limit for the tube straight leg (free-span) region of the tube for degradation over an unlimited axial extent, and for degradation over limited axial extent at the tube support plate and AVB intersections.

5.7.11 Evaluation of Tube Degradation

The original Kewaunee Model 51 steam generators used Alloy 600 mill-annealed tubing, partial depth mechanical roll expansion, and carbon steel tube support plates (TSPs) with drilled tube holes and separate drilled flow holes. The original Kewaunee steam generators experienced outer diameter stress corrosion cracking (ODSCC) in the tube-to-tubesheet crevice, denting at the top of tubesheet region with primary water stress corrosion cracking (PWSCC) in this area, ODSCC at TSP intersections, and tube wear at anti-vibration bar (AVB) intersections.

5.7.11.1 Input Parameters and Assumptions

Currently, the Kewaunee plant uses Westinghouse Model 54F RSGs. The Model 54F steam generator has 3,592 original Alloy 690 thermally treated tubes. The tube-to-tubesheet gap is closed by a hydraulic expansion process. The TSPs are constructed of 405 stainless steel, with quatrefoil design tube holes. The quatrefoil tube hole design allows for bulk fluid flow axially along the tube, therefore no interstitial flow holes are required. The row 1 U-bend minimum bend radius is 3.141 in., which is greater than the original steam generator row 1 U-bend minimum bend radius of 2.19 in. The larger bend radius reduces residual stresses from fabrication.

Additionally, the first nine rows of tubes received a supplemental thermal treatment of the U-bend region following bending. This thermal treatment should reduce the residual stresses from bending to near straight-leg residual-stress levels. Pre-operation in situ heat treatment of row 1 and 2 U-bend regions in plants with Alloy 600 mill-annealed tubing has precluded PWSCC initiation for up to 11.3 EFPY at operating temperatures of up to 618°F. The supplemental thermal treatment performed during manufacture is expected to provide an enhanced treatment compared to the in situ heat treatment performed in the field, prior to operation.

The AVB design includes tighter manufacturing tolerances and reduced AVB to tube-gap dimensions. Six AVBs are included in the Model 54F design versus four in the original steam generator design. An elevated feedring is used that helps prevent waterhammer events and feedwater nozzle cracking.

5.7.11.2 Description of Analysis/Evaluation

As the Kewaunee RSGs were installed in the Fall of 2001, the first in-service inspection of the RSGs has not been performed. The evaluation of the steam generator tubing performance is based on the accumulated operating history of advanced model steam generator designs (that is, Model D5, Model F, and Model 51F) that utilize similar design improvements, for the expected post-uprating hot-leg temperature.

5.7.11.3 Acceptance Criteria

Acceptability of the RSG design at the expected uprating temperature is based on expected stress corrosion cracking (SCC) resistance of the steam generator tubing material, and potential mechanical-degradation mechanisms (that is, AVB wear).

5.7.11.4 Results

The Model 54F design employs features that have historically been shown to provide significant design improvements in tubing SCC resistance. Plants with similar design features (that is, Model F design features) have operated for up to 15 EFPY at an equivalent temperature of 618°F with no confirmed reports of ODSCC or PWSCC degradation in domestically operated units. The first replacement units of this design (Surry) have operated since 1980 with no reports of SCC. It should be noted that in 1996 and 1997, reports of ODSCC and PWSCC in Alloy 600 thermally treated tubing were made at one plant with Model F steam generators. Significant difference of opinion regarding the validity of these indications has been made by various eddy current analysts who have reviewed this data. The tubes with these signals have been removed from service by plugging, and these tubes were not pulled for destructive examination. Reference 18 suggests that Alloy 690 thermally treated tubing material may be essentially immune to PWSCC mechanisms. The supplemental U-bend thermal treatment and hydraulic expansion process are attempts to further reduce PWSCC initiation potential.

Reference 18 presents an evaluation of the improvement in ODSCC resistance for Alloy 600 thermally treated and Alloy 690 thermally treated tubing compared to Alloy 600 mill-annealed tubing, as well as an evaluation of improvement in SCC resistance for the Model F design features. Reference 18 establishes lower-bound, median, and upper-bound corrosion estimates in terms of percentage of tubes plugged over 35 and 50 EFPY operating periods. Using the empirical operating history from plants with Alloy 600 mill-annealed tubing as a basis, normalized to a hot-leg temperature of 618°F, the 35 EFPY median corrosion estimates total <0.30-percent plugged for Alloy 600 thermally treated tubing.

The Alloy 690 thermally treated tubing performance is expected to be an improvement over the Alloy 600 thermally treated performance. The historical dominant ODSCC tube-degradation mechanism affecting plants with Alloy 600 mill-annealed tubing has been axial ODSCC at carbon steel, drilled hole TSP intersections. This mechanism has been addressed partly through the use of Alloy 690 thermally treated tube material, quatrefoil tube holes, stainless steel TSP material, and strict secondary side water chemistry control consistent with the EPRI Secondary Water Chemistry Guidelines. ODSCC at the top of tubesheet region is addressed partly through the use of Alloy 690 thermally treated tube material, hydraulic tube expansion, and strict secondary side water chemistry control consistent with the EPRI Secondary Water Chemistry Guidelines.

The effects of expansion process alone can be evidenced in tube degradation data from an operating plant with Model D4 steam generators. This plant contains Alloy 600 mill annealed tubing, with both mechanical roll expanded tubes and WEXTEX explosively expanded tubes. Through 8.99 EFPY at 621°F operating temperature, the ODSCC initiation rate for the mechanically roll expanded tubes is greater than 20 times the WEXTEX tube ODSCC initiation rate. Explosive expansion and hydraulic expansion residual stresses should be similar.

The current hot-leg-operating temperature of the Kewaunee plant is 599.1°F, while the expected hot-leg-operating temperature following the 7.4-percent uprating is 606.8°F. As the original Kewaunee steam generators were replaced with Model 54F steam generators in the Fall 2001, the first in-service inspection of the Kewaunee RSGs has not occurred. Return to power following steam generator replacement was approximately December 6, 2001. The first in-service inspection of the RSGs is scheduled for April 2003. Based on the Model F operating experience, with up to 15 EFPY with no confirmed SCC in domestic units at an operating temperature basis of 618°F, no SCC potential within the first 20.43 EFPY operational period (using an Arrhenius Equation and assumed ODSCC initiation activation energy of 35 kcal/mole) is anticipated in the Kewaunee RSGs for operation at the uprated-power conditions.

As the first in-service inspection of the Kewaunee RSGs has not occurred, no operational performance data is available. Similar units (such as Cook Unit 2, with Model 54F steam generators) have reported no SCC mechanisms and no AVB wear through approximately 7.27 EFPY. The only steam generator tube-degradation mechanisms reported at Cook Unit 2 are wear at TSP intersections (affecting only two tubes), and tube wear due to foreign object interaction (Reference 20).

Future steam generator tube degradation will be addressed through the condition-monitoring and operational-assessment process, governed by the EPRI Tube Integrity Guideline (Reference 19). The only tube-degradation mechanisms that may be affected in the near term by the 7.4-percent uprating are wear at TSP intersections and AVB wear. Any change in postulated TSP interaction wear rate or change in postulated AVB wear initiation and growth rate are expected to be negligible, based on the inspection histories to date for operating steam generators with similar design features. As the previously reported TSP wear indications at Cook Unit 2 did not change over the last operating cycle, no AVB wear has been reported to date, and the operating history from similar RSGs indicate extremely low-growth rates for these mechanisms, steam generator tube integrity is not expected to be impacted by the 7.4-percent uprating An analytical assessment of the impact of the uprated-power levels on tube-wear rates was performed. The results of that assessment, discussed earlier in this report, confirmed that there would be little change in wear rates as a result of the increase power level. The most recent Cook 2 eddy current inspection data indicates that no tubes were plugged due to wear depth exceeding the Technical Specification repair limit of 40 percent through wall by NDE. The growth rate of the reported TSP wear indications was 0.

5.7.11.5 Conclusion

Based on the design features inherent to the Model 54F steam generator, accumulated EFPY since replacement, and operating temperature following uprating, it is expected that tube plugging will be bounded by 0.1 percent at the end of current license. SCC mechanisms are not expected to be observed at the end of current license. The 0.1-percent plugging allowance is a conservative estimate based on potential tube plugging due to mechanisms such as foreign object wear, and AVB wear. Maintenance of secondary side water chemistry within guidelines established by the EPRI Secondary Water Chemistry Guidelines should further reduce the potential for SCC mechanisms to affect steam generator operability.

5.7.11.6 References

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+a,c

Table 5.7-1

Kewaunee 7.4-% Power Uprate: Results of Thermal-Hydraulic Evaluations

+a,c

Table 5.7-2

Feedwater Nozzle and Thermal Sleeve
Maximum Primary-plus-Secondary Stress Intensity Ranges
(stratification combinations at hot-side locations)

+a,c

Table 5.7-2 (continued)

Feedwater Nozzle and Thermal Sleeve
Maximum Primary-plus-Secondary Stress Intensity Ranges
(stratification combinations at hot-side locations)

+a,c

Table 5.7-3

Feedwater Nozzle and Thermal Sleeve
Maximum Primary-plus-Secondary Stress Intensity Ranges
(stratification combinations at cold-side locations)

+a,c

Table 5.7-3 (continued)

Feedwater Nozzle and Thermal Sleeve
Maximum Primary-plus-Secondary Stress Intensity Ranges
(stratification combinations at cold-side locations)

+a,c

Table 5.7-4

J-Nozzle-to-Feedring Weld
Maximum Stress Intensity Ranges at Limiting Locations

Table 5.7-5 Summary of Full-Power Operating Conditions[1]		
Parameter	High T_{ave} Conditions	Low T_{ave} Conditions
P_{pnm} (psia)	2250.0	2250.0
P_{sec} (psia)	747.0	645.0
T_{steam} (°F)	510.4	494.0
T_{hot} (°F)	606.8	590.8
No Load Temp. (°F)	547.0	547.0

Note

1. Corresponds to 10-% tube plugging

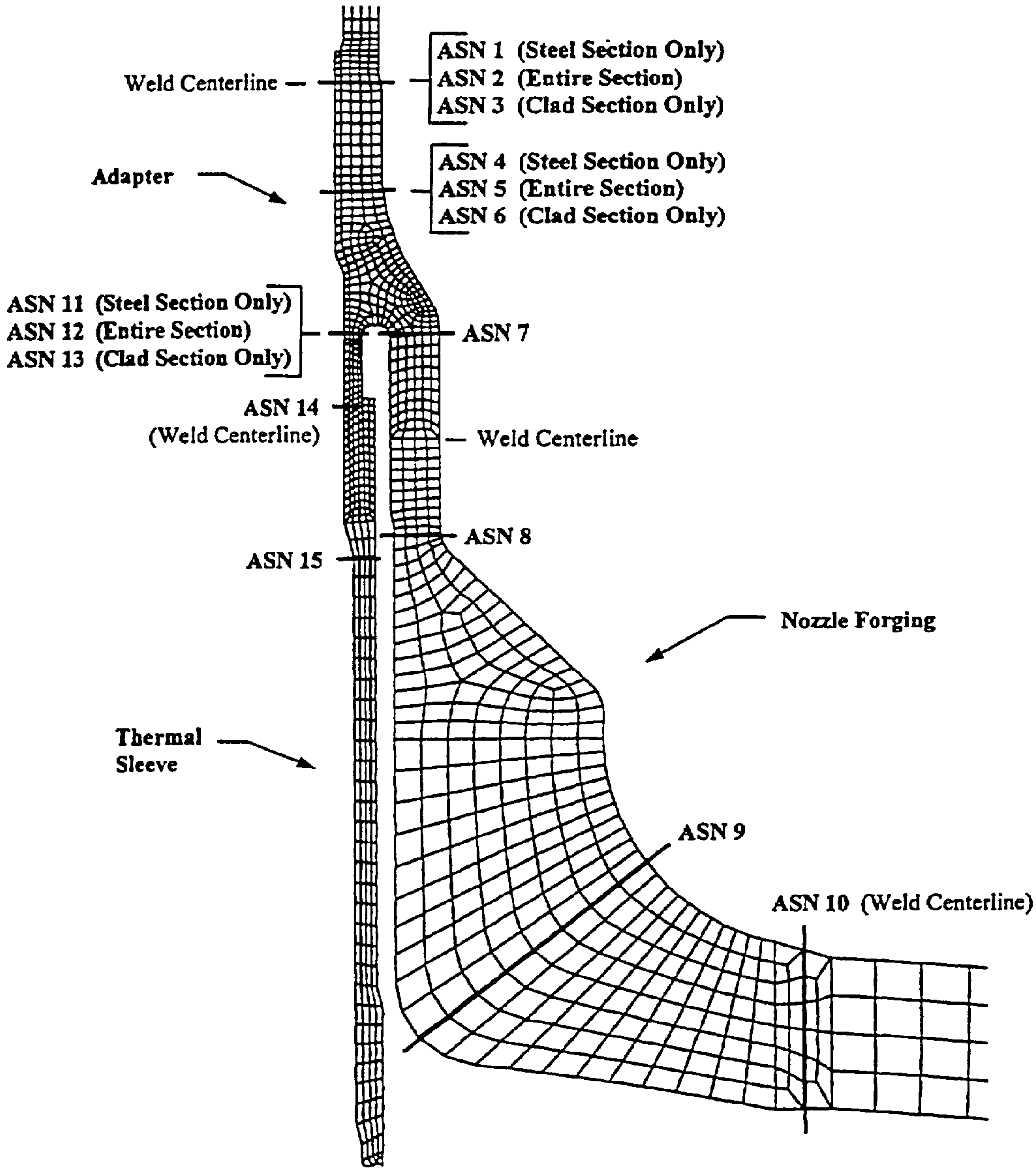

Figure 5.7-1

Limiting Stress Locations (ASNs) in Feedwater Nozzle and Thermal Sleeve

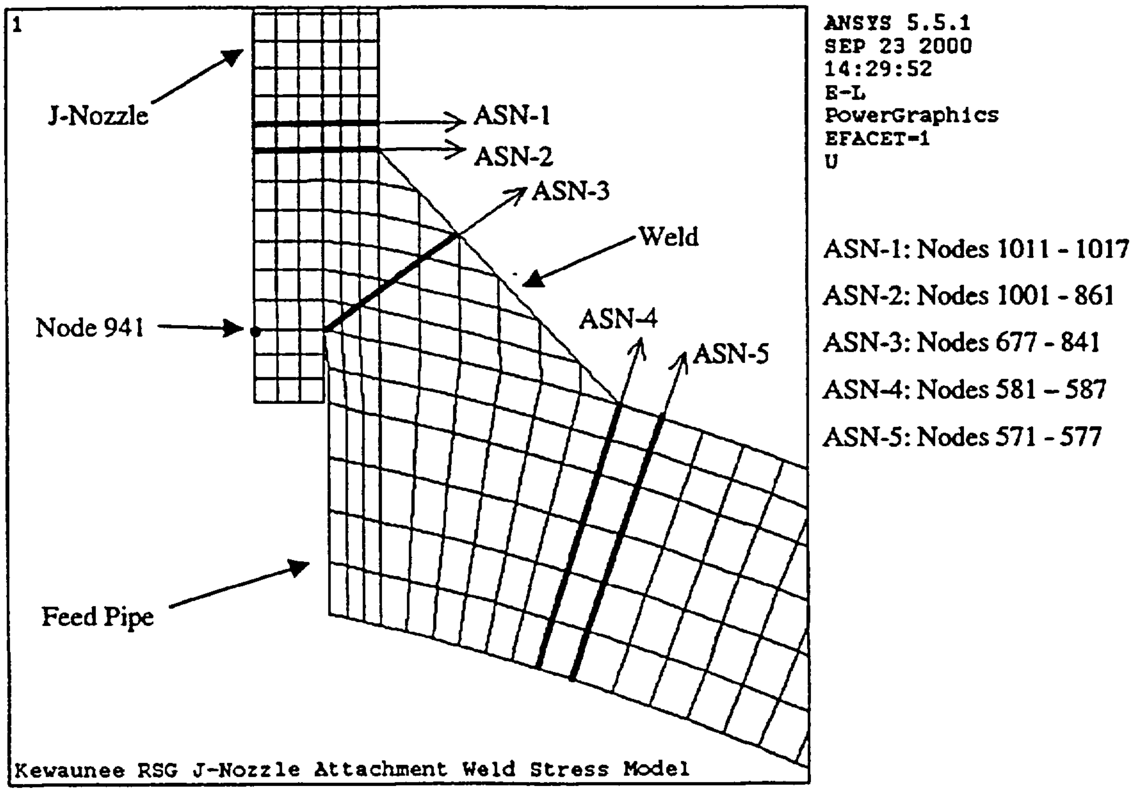

Figure 5.7-2

Limiting Stress Locations (ASNs) in J-Nozzle-to-Feedring Weld

(inside surface of ASN-3 at weld root produces limiting fatigue usage factor)

5.8 Pressurizer Component Evaluations

Kewaunee Nuclear Power Plant (KNPP) has proposed uprating their operating nuclear steam supply system (NSSS) power level from 1,657.1 MWt (828.6 MWt/loop) to 1,780 MWt (890 MWt/loop). This represents a power uprating of 7.4 percent. To support the planned 7.4-percent power uprating, the pressurizer has been evaluated for operation at the uprated power conditions. Any pressurizer design transients that were affected by the upgraded power levels were addressed in the evaluations.

5.8.1 Pressurizer Evaluation

5.8.1.1 Introduction

The functions of the pressurizer are to absorb any expansion or contraction of the primary reactor coolant due to changes in temperature and/or pressure and, in conjunction with the pressure control system components, to keep the Reactor Coolant System (RCS) at the desired pressure. Since the pressurizer is connected to the RCS at the hot leg of one of the reactor coolant loops (RCLs), this allows for inflow to, or outflow from, the pressurizer as required. The first function is accomplished by keeping the pressurizer approximately half-full of water and half-full of steam at normal conditions. The second function is accomplished by keeping the temperature in the pressurizer at the water saturation temperature (T_{sat}) corresponding to the desired pressure. The temperature of the water and steam in the pressurizer can be raised by operating electric heaters at the bottom of the pressurizer and can be lowered by introducing relatively cool spray water into the steam space at the top of the pressurizer.

The components in the lower end of the pressurizer (such as the surge nozzle, lower head/heater well, and support skirt) are affected by pressure and surges through the surge nozzle. The components in the upper end of the pressurizer (such as the spray nozzle, safety and relief nozzle, upper head/upper shell, manway, and instrument nozzle) are affected by pressure, spray flow through the spray nozzle, and steam temperature differences.

5.8.1.2 Input Parameters and Assumptions

The limiting operating conditions of the pressurizer occur when the RCS pressure is high and the RCS hot leg (T_{hot}) and cold leg (T_{cold}) temperatures are low. This maximizes the ΔT that is experienced by the pressurizer. Due to flow out of and into the pressurizer during various transients, the surge nozzle alternately sees water at the pressurizer temperature (T_{sat}) and water from the RCS hot leg at T_{hot}. If the RCS pressure is high (which means, correspondingly, that T_{sat} is high) and T_{hot} is low, then the surge nozzle will see maximum thermal gradients (ΔT_{hot} = temperature difference between T_{hot} and the pressurizer [surge nozzle] temperature); and thus experience the maximum thermal stress. Likewise, the spray nozzle and upper shell temperatures alternate between steam at T_{sat} and spray water, which, for many transients, is at T_{cold}. Thus, if RCS pressure is high (T_{sat} is high) and T_{cold} is low, then the spray nozzle and upper shell will also experience the maximum thermal gradients (ΔT_{cold} = temperature difference between T_{cold} and the pressurizer [spray nozzle] temperature) and thermal stresses. The summary of ΔT hot and cold values can be seen in Table 5.8-1.

The KNPP plant and the Point Beach plant have very similar pressurizer units. Both pressurizers were built to the same base design specification, and the units share the same original design basis analysis as documented in the plant-specific stress reports. The Performance Capability Working Group (PCWG) T_{hot} and T_{cold} parameters applicable for the Point Beach RSG (Replacement Steam Generator) analysis and those applicable for the Kewaunee RSG and uprate conditions are very similar and any difference between the values would have had a negligible effect on the present evaluation. On this basis it was possible to utilize both the Point Beach and the Kewaunee pressurizer design basis analyses as the basis for the uprated power evaluation.

5.8.1.3 Description of Analysis/Evaluation

The two components to be reanalyzed were the spray nozzle and the surge nozzle. In both cases a prior fatigue analysis was adjusted to reflect any changes in the ΔT or cycles for a particular transient. This ΔT_{cold} or ΔT_{hot} at which the pressurizer was previously analyzed was compared to the ΔT calculated from the uprate parameters. By evaluating the surge and spray nozzle, which are the most highly stressed components, all other components are qualified.

The PCWG uprate parameters were considered in this uprated power evaluation. No other changes are considered to the pressure or other thermal-hydraulic design parameters for the 7.4-percent Power Uprate Project, since the NSSS design transients applicable to the uprated power conditions related to the pressurizer have not changed from those applicable for the RSG Program.

In order to consider the effects of the change in design parameters an existing analysis was used as a basis for the spray nozzle part of the current analysis. Since the design basis used in the technical report for Point Beach references the same generic stress reports as referenced by the Kewaunee site specific stress report, as well as the same generic E – Spec. the two plant sites were considered similar enough so that the Point Beach analysis can be used as a basis for the 7.4% uprate analysis.

The Point Beach analysis stated that in review of the uprating design transients and the original analyses (i.e., the original generic stress reports) that the ΔT parameters would not have had any significant impact on Sections 3.8 – 3.14 (which list various pressurizer components) of the generic stress reports. By comparing the Kewaunee ΔT for the spray nozzle (ΔT_{cold}) as well as the cycles for each transient, to that same information for Point Beach, it can be concluded that the above statement on significant impact to various pressurizer components can be applied to the Kewaunee pressurizer for the RSG program as well.

For the other components (except the spray nozzle) and analyses (Safety and Relief Nozzle, Generic Seismic, Lower Head, Heater Well, Upper Head and Shell) only an increase in the number of cycles for the Leak Test transient was considered resulting in an increase in the cumulative fatigue usage factor for those components. However, for the Kewaunee RSG program, the number of cycles for this transient did not change from the original design basis; therefore, those components did not have to be reanalyzed. For the Support Skirt/ Flange, any change in the ΔT_{hot} from the surge nozzle transients would have only a negligible impact for this particular component, and it was therefore not reanalyzed.

For the surge nozzle, an analysis was done to modify the cumulative fatigue usage factor for the Kewaunee surge nozzle due to thermal stratification. The original design basis for Kewaunee, provided the input ΔT and number of cycles for each transient used in the surge nozzle stress report. Since the number of cycles in the thermal analysis was greater than or equal to the number of cycles in the RSG uprate program the cycles were not changed for this analysis. However, the ΔT for some of the transient groups did change, and were modified using a temperature ratio in order to take into account the new ΔTs.

5.8.1.4 Acceptance Criteria

The cumulative fatigue usage factor calculated by using the ΔT values as shown in Table 5.8-1 for both critical components remains under one, and component stresses satisfy ASME Code, Section III stress allowables (Reference 1).

5.8.1.5 Results

Table 5.8-1 compares the operating conditions' change in temperature (ΔT) values for both the Thot and Tcold parameters. Table 5.8-2 compares the revised fatigue usage factors for various components, with those calculated previously. Table 5.8-3 compares the original and revised stress intensity ranges for uprate, compared to the ASME Code limit. For the surge nozzle the uprate did not affect the maximum range stress intensity. For the spray nozzle the uprate did not affect the maximum range stress intensity either. However, since the maximum range stress intensity exceeds the ASME Code limit, a simplified elastic – plastic analysis was done in accordance with Section NB – 3228.3 of the ASME Code. The resultant stress intensity was less than the ASME Code limit, and therefore acceptable.

5.8.1.6 Conclusions

As can be seen by Table 5.8-2, the analysis of the Kewaunee pressurizer for the plant operation conditions that applied for the RSG installation, verified that the pressurizer is qualified for these operating conditions.

A comparison of the fatigue usage values for the components clearly shows that the spray nozzle and the surge nozzle are limiting for this pressurizer. Demonstrating the acceptability of both the spray and surge nozzles therefore proves that the remaining pressurizer components remain qualified as well. Both stress levels and cumulative fatigue usage were shown to be below the applicable ASME Code (Reference 1) limits (fatigue usage < 1.0) for both the spray and surge nozzles.

A comparison in Table 5.8-1 of the 7.4-percent uprated power operating conditions (the ΔTs) to those used for the RSG evaluations as discussed in Section 5.8.1.3 showed that the RSG conditions enveloped the uprated power conditions for the pressurizer. On this basis, the pressurizer is qualified for operation at the 7.4 percent uprated power level.

5.8.1.7 References

1. "Rules for Construction of Nuclear Power Plant Components," ASME Boiler and Pressure Vessel Code Section III, 1965 Edition, Summer 1966 Addenda, The American Society of Mechanical Engineers, New York, New York, USA.

Table 5.8-1

Summary of Change in Temperature
(ΔT – surge and spray nozzle to loop parameter)

Component	Parameters	ΔT (°F) Original Design Basis	RSG		7.4-% Uprate	
			Temp (°F) PCWG	ΔT (°F)	Temp (°F) PCWG	ΔT (°F)
Surge Nozzle	T_{hot}	125.0	586.3	125.0	590.8	62.2
Spray Nozzle	T_{cold}	125.0	521.9	160.0	521.9	132.1
Reference No.		4	10	10	8	8

+a,c

Table 5.8-2 Kewaunee Fatigue Usage Comparisons (ASME Code allowable < 1.0)		

Table 5.8-3 Maximum Primary-plus-Secondary Stress Intensity Ranges			

ATTACHMENT 10

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

Dated

January 13, 2003

License Amendment Request 193

Strike-Out Pages for License, Technical Specifications, and Bases

Operating License, page 3
TS iv
TS vi
TS 1.0-4
TS 3.1-6
TS B3.1-6
TS B3.1-7
Figure TS 3.1-1
Figure TS 3.1-2
TS 6.9-3
TS 6.9-4
TS 6.9-5

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR, Chapter 1: (1) Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Section 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70, (2) is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect, and (3) is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The NMC is authorized to operate the facility at steady-state reactor core power levels not in excess of ~~1650~~ 1673 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. ~~164~~ are hereby incorporated in the license. The NMC shall operate the facility in accordance with the Technical Specifications.

(3) Fire Protection

The NMC shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the KNPP Fire Plan, and as referenced in the Updated Safety Analysis Report, and as approved in the Safety Evaluation Reports, dated November 25, 1977, and December 12, 1978 (and supplement dated February 13, 1981) subject to the following provision:

The NMC may make changes to the approved Fire Protection Program without prior approval of the Commission, only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(4) Physical Protection

The NMC shall fully implement and maintain in effect all provisions of the Commission-approved "Kewaunee Nuclear Power Plant Security Manual," Rev. 1, approved by the NRC on December 15, 1989, the "Kewaunee Nuclear Power Plant Security Force Training and Qualification Manual," Rev. 7, approved by the NRC on November 17, 1987, and the "Kewaunee Nuclear Power Plant Security Contingency Plan," Rev. 1, approved by the NRC on September 1, 1983. These manuals include amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p).

(5) Fuel Burnup

The maximum rod average burnup for any rod shall be limited to 60 GWD/MTU until completion of an NRC environmental assessment supporting an increased limit.

LIST OF FIGURES

Note:

[1] ~~Although the~~ The curves ~~were developed for 33 EFPY, they~~ are limited to ~~28~~ 31.1 EFPY ~~(corresponding to the end of cycle 28) by WPSC Letter NRC-99-017.~~ due to changes in vessel fluence associated with operation at uprated power.

j. MODES

MODE	REACTIVITY $\Delta k/k$	COOLANT TEMP T_{avg} °F	FISSION POWER %
REFUELING	$\leq$ -5%	$\leq$ 140	~0
COLD SHUTDOWN	$\leq$ -1%	$\leq$ 200	~0
INTERMEDIATE SHUTDOWN	(1)	> 200 < 540	~0
HOT SHUTDOWN	(1)	$\geq$ 540	~0
HOT STANDBY	< 0.25%	~T_{oper}	< 2
OPERATING	< 0.25%	~T_{oper}	$\geq$ 2
LOW POWER PHYSICS TESTING	(To be specified by specific tests)		
(1) Refer to the required SHUTDOWN MARGIN as specified in the Core Operating Limits Report.			

k. REACTOR CRITICAL

The reactor is said to be critical when the neutron chain reaction is self-sustaining.

l. REFUELING OPERATION

REFUELING OPERATION is any operation involving movement of reactor vessel internal components (those that could affect the reactivity of the core) within the containment when the vessel head is unbolted or removed.

m. RATED POWER

RATED POWER is the steady-state reactor core output of ~~1,650~~1,673 MWt.

n. REPORTABLE EVENT

A REPORTABLE EVENT is defined as any of those conditions specified in 10 CFR 50.73.

b. Heatup and Cooldown Limit Curves for Normal Operation

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures TS 3.1-1 and TS 3.1-2. Figures TS 3.1-1 and TS 3.1-2 are applicable for the service period of up to 33[(1)] effective full-power years.

   A. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.

   B. Figures TS 3.1-1 and TS 3.1-2 define limits to assure prevention of non-ductile failure only. For normal operation other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

   C. The isothermal curve in Figure TS 3.1-2 defines limits to assure prevention of non-ductile failure applicable to low temperature overpressurization events only. Application of this curve is limited to evaluation of LTOP events whenever one or more of the RCS cold leg temperatures are less than or equal to the LTOP enabling temperature of 200°F.

2. The secondary side of the steam generator must not be pressurized > 200 psig if the temperature of the steam generator is < 70°F.

3. The pressurizer cooldown and heatup rates shall not exceed 200°F/hr and 100°F/hr, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is > 320°F.

4. The overpressure protection system for low temperature operation shall be operable whenever one or more of the RCS cold leg temperatures are ≤ 200°F, and the reactor vessel head is installed. The system shall be considered operable when at least one of the following conditions is satisfied:

   A. The overpressure relief valve on the Residual Heat Removal System (RHR 33-1) shall have a set pressure of ≤ 500 psig and shall be aligned to the RCS by maintaining valves RHR 1A, 1B, 2A, and 2B open.

      1. With one flow path inoperable, the valves in the parallel flow path shall be verified open with the associated motor breakers for the valves locked in the off position. Restore the inoperable flow path within 5 days or complete depressurization and venting of the RCS through a ≥ 6.4 square inch vent within an additional 8 hours.

      2. With both flow paths or RHR 33-1 inoperable, complete depressurization and venting of the RCS through at least a 6.4 square inch vent pathway within 8 hours.

[(1)] ~~Although the~~ The curves ~~were developed for 33 EFPY, they~~ are limited to ~~28~~ 31.1 EFPY ~~(corresponding to the end of cycle 28) by WPSC Letter NRC-99-017.~~ due to changes in vessel fluence associated with operation at uprated power.

Limit curves for normal heatup and cooldown of the primary Reactor Coolant System have been calculated using the methods discussed above and limited application to ASME Boiler and Pressure Vessel Code Case N-588 to the circumferential beltline weld. The derivation of the limit curves is consistent with the NRC Regulatory Standard Review Plan(8) and Footnote.(9)

Transition temperature shifts occurring in the pressure vessel materials due to radiation exposure have been obtained directly from the reactor pressure vessel surveillance program. As presented in WCAP 14279, Revision 1,(10) weld metal Charpy test specimens from Capsule S indicate that the core region weld metal exhibits the largest shift in RT_{NDT} (250°F).

The results of Irradiation Capsules V, R, P, and S analyses are presented in WCAP 8908,(11) WCAP 9878,(12) WCAP-12020,(13) WCAP-14279,(14) and WCAP-14279, Revision 1(10) respectively. Heatup and cooldown limit curves for normal operation of the reactor vessel are presented in Figures TS 3.1-1 and TS 3.1-2 and represent an operational time period of 33[1] effective full-power years.

The isothermal cooldown limit curve (Figure TS 3.1-2) is used for evaluation of low temperature overpressure protection (LTOP) events. This curve is applicable for 33[1] effective full-power years of fluence (through the end of OPERATING cycle 33[1]). If a low temperature overpressure event occurred, the RCS pressure transient would be evaluated to the limits of this figure to verify the integrity of the reactor vessel. If these limits are not exceeded, vessel integrity is assured and a TS violation has not occurred.

Note:

[1] ~~Although the~~ The curves ~~were developed for 33 EFPY, they~~ are limited to ~~28~~ 31.1 EFPY ~~(corresponding to the end of cycle 28) by WPSC Letter NRC-99-017.~~ due to changes in vessel fluence associated with operation at uprated power.

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(8) "Fracture Toughness Requirements," Branch Technical Position MTEB 5-2, Chapter 5.3.2 in Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, NUREG-0800, 1981.

(9) 1989 ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, Appendix G, "Fracture Toughness Criteria for Protection Against Failure."

(10) C. Kim, et al., "Evaluation of Capsule S from the Kewaunee and Capsule A35 from the Maine Yankee Nuclear Power Reactor Vessel Radiation Surveillance Programs," WCAP-14279, Revision 1, September 1998.

(11) S.E. Yanichko, S. L. Anderson, and K. V. Scott, "Analysis of Capsule V from the Wisconsin Public Service Corporation Kewaunee Nuclear Plant Reactor Vessel Radiation Surveillance Program," WCAP 8908, January 1977.

(12) S.E. Yanichko, et al., "Analysis of Capsule R from the Wisconsin Public Service Corporation Kewaunee Nuclear Plant Reactor Vessel Radiation Surveillance Program," WCAP 9878, March 1981.

(13) S.E. Yanichko, et al., "Analysis of Capsule P from the Wisconsin Public Service Corporation Kewaunee Nuclear Power Plant Reactor Vessel Radiation Surveillance Program," WCAP-12020, November 1988.

(14) E. Terek, et al., "Analysis of Capsule S from the Wisconsin Public Service Corporation Kewaunee Nuclear Power Plant Reactor Vessel Radiation Surveillance Program," WCAP-14279, March 1995.

Pressurizer Limits (TS 3.1.b.3)

Although the pressurizer operates at temperature ranges above those for which there is reason for concern about brittle fracture, OPERATING limits are provided to ensure compatibility of operation with the fatigue analysis performed in accordance with Code requirements. In-plant testing and calculations have shown that a pressurizer heatup rate of 100°F/hr cannot be achieved with the installed equipment.

Low Temperature Overpressure Protection (TS 3.1.b.4)

The Low Temperature Overpressure Protection System must be OPERABLE during startup and shutdown conditions below the enable temperature (i.e., low temperature) as defined in Branch Technical Position RSB 5-2 as modified by ASME Boiler and Pressure Vessel Code Case N–514. Based on the Kewaunee Appendix G LTOP protection pressure-temperature limits calculated through 33[1] effective full-power years, the LTOP System must be OPERABLE whenever one or more of the RCS cold leg temperatures are ≤ 200°F and the head is on the reactor vessel. The LTOP system is considered OPERABLE when all four valves on the RHR suction piping (valves RHR-1A, 1B, 2A, 2B) are open and valve RHR-33-1, the LTOP valve, is able to relieve RCS overpressure events without violating Figure TS 3.1-2.

The set pressure specified in TS 3.1.b.4 includes consideration for the opening pressure tolerance of ± 3% (± 15 psig) as defined in ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NC: Class 2 Components for Safety Relief Valves. The analysis of pressure transient conditions has demonstrated acceptable relieving capability at the upper tolerance limit of 515 psig.

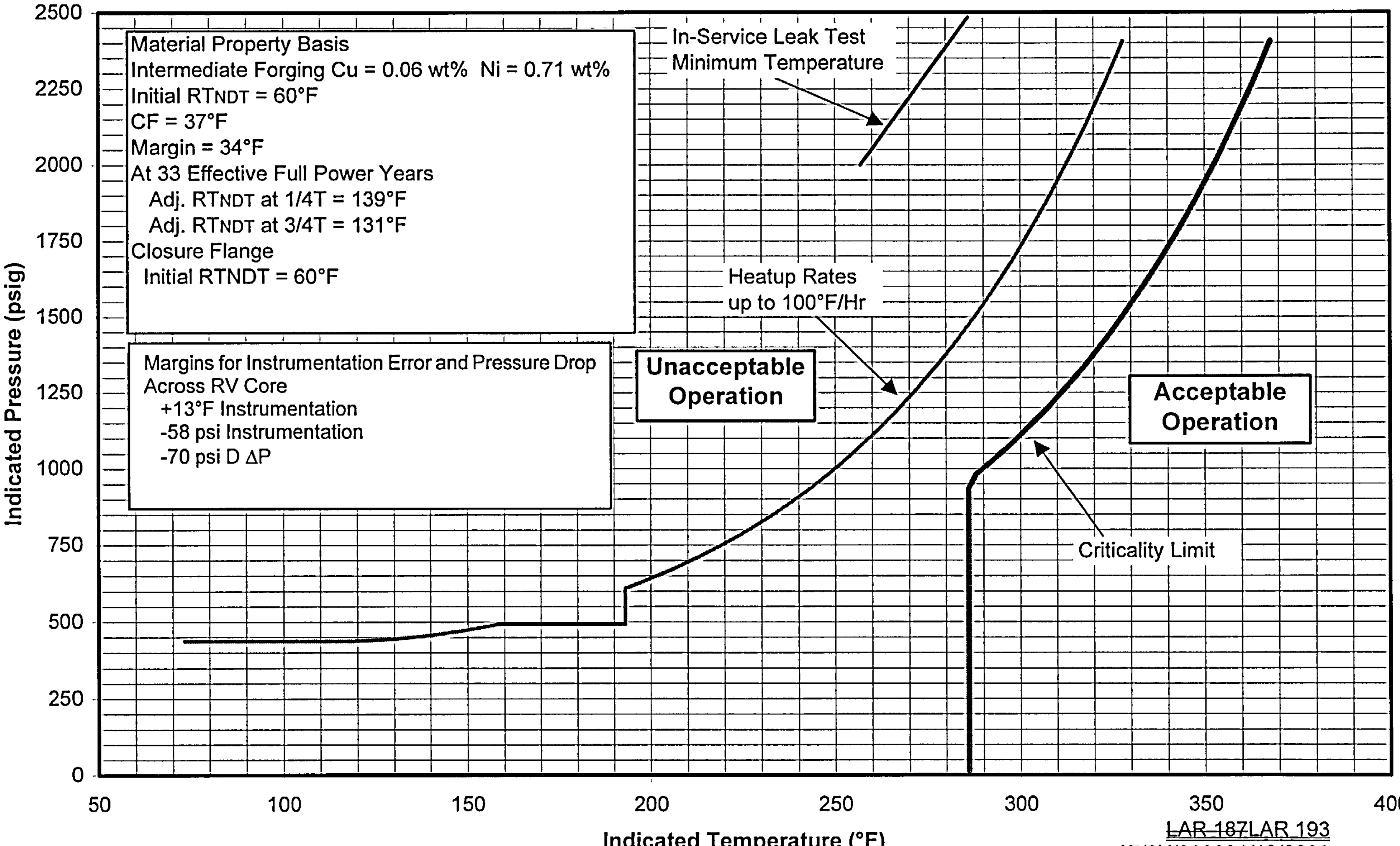
If one train of RHR suction piping to RHR 33-1 is isolated, then the valves and valve breakers in the other train shall be verified open, and the isolated flowpath must be restored within five days. If the isolated flowpath cannot be restored within five days, then the RCS must be depressurized and vented through at least a 6.4 square inch vent within an additional eight hours.

If both trains of RHR suction are isolated or valve RHR 33-1 is inoperable, then the system can still be considered OPERABLE if an alternate vent path is provided which has the same or greater effective flow cross section as the LTOP safety valve (≥ 6.4 square inches). If vent path is provided by physical openings in the RCS pressure boundary (e.g., removal of pressurizer safety valves or steam generator manways),then the vent path is considered secured in the open position.

Note

[1] ~~Although the~~ The curves ~~were developed for 33 EFPY, they~~ are limited to ~~28~~ 31.1 EFPY ~~(corresponding to the end of cycle 28) by WPSC Letter NRC-99-017.~~ due to changes in vessel fluence associated with operation at uprated power.

FIGURE TS 3.1-1
KEWAUNEE UNIT NO. 1 HEATUP LIMITATION CURVES
APPLICABLE FOR PERIODS UP TO 33[1] EFFECTIVE FULL-POWER YEARS



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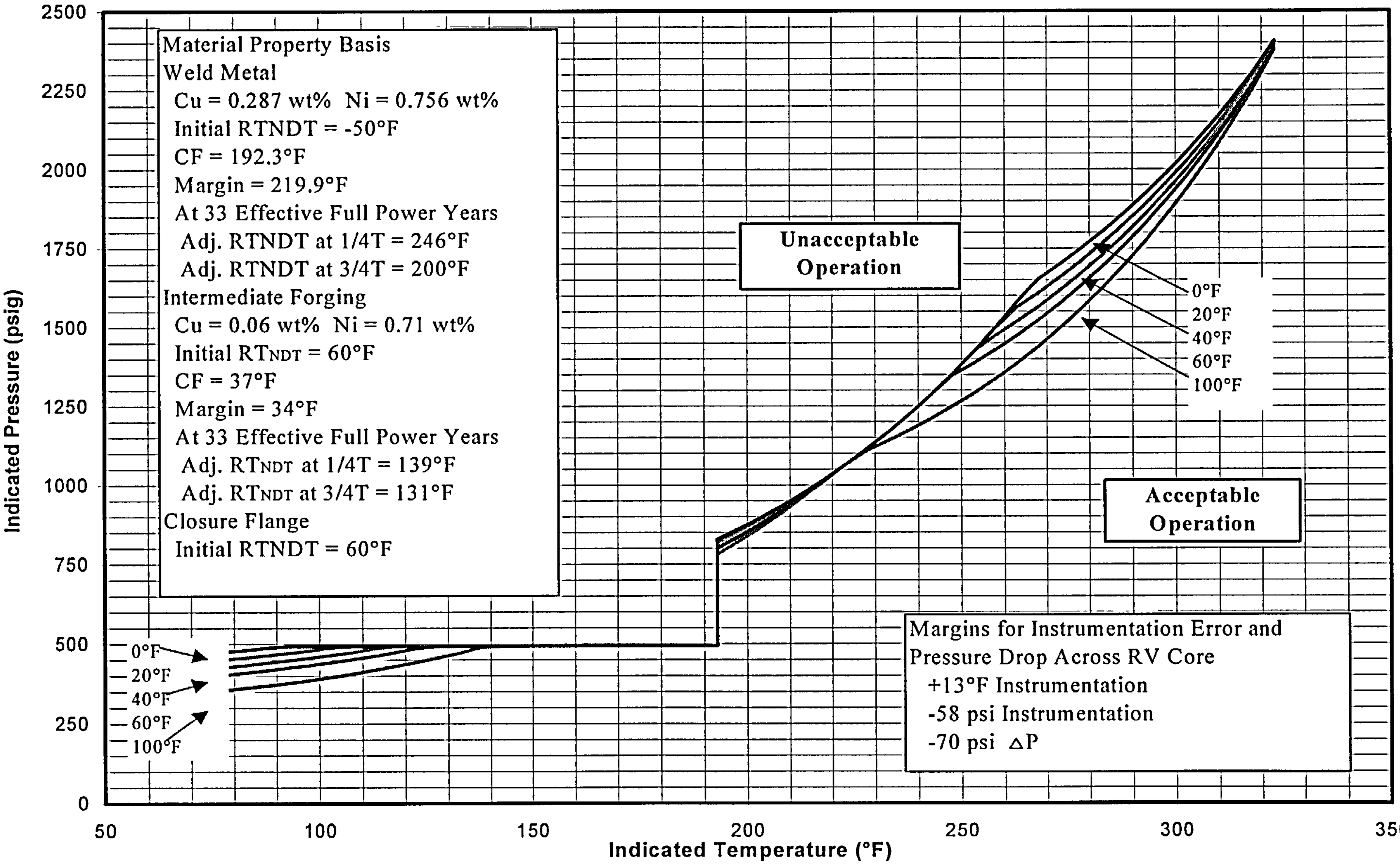
~~07/2X/2002~~01/13/2003

NOTE:
[1] ~~Although the~~The curves ~~were developed for 33 EFPY, they~~ are limited to ~~28~~31.1 EFPY due to changes in vessel fluence associated with operation at uprated power. ~~(corresponding to the end of cycle 28) by WPSC Letter Nuclear Regulatory Commission99-017~~

FIGURE TS 3.1-2
KEWAUNEE UNIT NO. 1 COOLDOWN LIMITATION CURVES
APPLICABLE FOR PERIODS UP TO 33 [1] EFFECTIVE FULL-POWER YEARS



[1] ~~Although the~~The curves ~~were developed for 33 EFPY, they~~ are limited to ~~28~~ 31.1 EFPY
LAR 187
due to changes in vessel fluence associated with operation at uprated powe.~~(corresponding to the end of cycle 28) by WPSC Letter NRC-99-017~~07/2X/2002 01/13/2003

3. Monthly OPERATING Report

Routine reports of OPERATING statistics and shutdown experience shall be submitted on a monthly basis to the Document Control Desk, U.S. Nuclear Regulatory Commission, Washington, D.C., 20555, with a copy to the appropriate Regional Office, to be submitted by the fifteenth of each month following the calendar month covered by the report.

4. Core Operating Limits Report (COLR)

A. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

(1)	TS 2.1	Reactor Core Safety Limit
(2)	TS 2.3.a.3.A	Overtemperature ΔT Setpoint
(3)	TS 2.3.a.3.B	Overpower ΔT Setpoint
(4)	TS 3.1.f.3	Moderator Temperature Coefficient (MTC)
(5)	TS 3.1.f.4	Moderator Temperature Coefficient (MTC)
(6)	TS 3.8.a.5	Refueling Boron Concentration
(7)	TS 3.10.a	Shutdown Margin
(8)	TS 3.10.b.1.A	$F_Q^N(Z)$ Limits
(9)	TS 3.10.b.1.B	$F_{\Delta H}^N$ Limits
(10)	TS 3.10.b.4	$F_Q^{EQ}(Z)$ Limits
(11)	TS 3.10.b.5.C.i	$F_Q^{EQ}(Z)$ penalty
(12)	TS 3.10.b.9	Axial Flux Difference Target Band
(13)	TS 3.10.b.11.A	Axial Flux Difference Envelope
(14)	TS 3.10.d.1	Shutdown Bank Insertion Limits
(15)	TS 3.10.d.2	Control Bank Insertion Limits
(16)	TS 3.10.k	Core Average Temperature
(17)	TS 3.10.l	Reactor Coolant System Pressure
(18)	TS 3.10.m.1	Reactor Coolant Flow

B. ~~The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:~~"The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC> When an initial assumed power level of 102% of the original rated power is specified in a previously approved method, 100.6% of uprated power may be used only when the main feedwater flow measurement (used as the input for reactor thermal output) is provided by the Crossflow untrasonic flow measurement system (Crossflow system) as described in report (15) listed below. When main feedwater flow measurements from the Crossflow System are unavailable, a power measurement uncertainty consistent with the instrumentation used shall be applied.

"Future revisions of approved analytical methods listed in this Technical Specification that currently reference the original Appendix K uncertainty of 102% of the original rated power should include the condition given above allowing use of 100.6% of uprated power in the safety analysis methodology when the Crossflow system is used for main feedwater flow measurement.

"The approved analytical methods are described in the following documents.

(1) SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION ON "QUALIFICATIONS OF REACTOR PHYSICS METHODS FOR APPLICATION TO KEWAUNEE" REPORT, dated August 21, 1979, report date September 29, 1978

(2) KEWAUNEE NUCLEAR POWER PLANT – REVIEW FOR KEWAUNEE RELOAD SAFETY EVALUATION METHODS TOPICAL REPORT WPSRSEM-NP, REVISION 3 (TAC NO MB0306) dated September 10, 2001.

(3) Nissley, M.E. et, al., "Westinghouse Large-Break LOCA Best-Estimate Methodology," WCAP-10924-P-A, Volume 1, Revision 1, Addendum 4, March 1991, Volume 1: Model Description and Validation; Addendum 4: Model Revisions.

(4) N. Lee et al., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A (Proprietary) and WCAP-10081-NP-A (Non-Proprietary), dated August 1985.

(5) C.M. Thompson, et al., "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condesation Model," WCAP-10054-P-A, Addendum 2, Revision 1 (Proprietary) and WCAP-10081-NP (Non-Proprietary), dated July 1997.

(6) XN-NF-82-06 (P)(A) Revision 1 and Supplements 2, 4, and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup, Exxon Nuclear Company, dated October 1986.

(7) ANF-88-133 (P)(A) and Supplement 1, "Qualification of Advanced Nuclear Fuels' PWR Design Meghodology for Rod Burnups of 62 GWd/MTU," Advanced Nuclear Fuels Corporation, dated December 1991.

(8) EMF-92-116 (P)(A) Revision 0, "Generic Mechanical Design Criteria for PWR Fuel Designs," Siemens Power Corporation, dated February 1999.

(9) XN-NF-77-57, Exxon Nuclear Power Distribution Control for Pressurized Water Reactors, Phase II, dated January 1978, and Supplement 2, dated October 1981.

(10) WCAP-10216-P-A, Rev. 1A, "Relaxation of Constant Axial Offset Control FQ Surveillance Technical Specification," February 1994 (W Proprietary).

(11) WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary)

(12) WCAP-8745-P-A, Design Bases for the Thermal Overtemperature ΔT and Thermal Overpower ΔT trip functions, September 1986.

(13) WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES-TOPICAL REPORT," September 1974 (Westinghouse Proprietary).

(14) WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," April 1995 (Westinghouse Proprietary).

~~(15) CENP-397-P-A, "Improved Flow Measurement Accuracy Using Cross Flow Ultrasonic Flow Measurement Technology, "Rev. 1, May 2000.~~

C. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

D. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

ATTACHMENT 11

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

Dated

January 13, 2003

<u>License Amendment Request 193</u>

Revised Pages for License, Technical Specifications, and Bases

Operating License, page 3
TS iv
TS vi
TS 1.0-4
TS 3.1-6
TS B3.1-6
TS B3.1-7
Figure TS 3.1-1
Figure TS 3.1-2
TS 6.9-3
TS 6.9-4
TS 6.9-5
TS 6.9-6

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR, Chapter 1: (1) Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Section 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70, (2) is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect, and (3) is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The NMC is authorized to operate the facility at steady-state reactor core power levels not in excess of 1673 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. are hereby incorporated in the license. The NMC shall operate the facility in accordance with the Technical Specifications.

(3) Fire Protection

The NMC shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the KNPP Fire Plan, and as referenced in the Updated Safety Analysis Report, and as approved in the Safety Evaluation Reports, dated November 25, 1977, and December 12, 1978 (and supplement dated February 13, 1981) subject to the following provision:

The NMC may make changes to the approved Fire Protection Program without prior approval of the Commission, only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(4) Physical Protection

The NMC shall fully implement and maintain in effect all provisions of the Commission-approved "Kewaunee Nuclear Power Plant Security Manual," Rev. 1, approved by the NRC on December 15, 1989, the "Kewaunee Nuclear Power Plant Security Force Training and Qualification Manual," Rev. 7, approved by the NRC on November 17, 1987, and the "Kewaunee Nuclear Power Plant Security Contingency Plan," Rev. 1, approved by the NRC on September 1, 1983. These manuals include amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p).

(5) Fuel Burnup

The maximum rod average burnup for any rod shall be limited to 60 GWD/MTU until completion of an NRC environmental assessment supporting an increased limit.

Section	Title	Page

LIST OF FIGURES

Note:

[1] The curves are limited to 31.1 EFPY due to changes in vessel fluence associated with operation at uprated power.

j. MODES

MODE	REACTIVITY $\Delta k/k$	COOLANT TEMP T_{avg} °F	FISSION POWER %
REFUELING	$\leq$ -5%	$\leq$ 140	~0
COLD SHUTDOWN	$\leq$ -1%	$\leq$ 200	~0
INTERMEDIATE SHUTDOWN	(1)	> 200 < 540	~0
HOT SHUTDOWN	(1)	$\geq$ 540	~0
HOT STANDBY	< 0.25%	~T_{oper}	< 2
OPERATING	< 0.25%	~T_{oper}	$\geq$ 2
LOW POWER PHYSICS TESTING	(To be specified by specific tests)		
(1) Refer to the required SHUTDOWN MARGIN as specified in the Core Operating Limits Report.			

k. REACTOR CRITICAL

The reactor is said to be critical when the neutron chain reaction is self-sustaining.

l. REFUELING OPERATION

REFUELING OPERATION is any operation involving movement of reactor vessel internal components (those that could affect the reactivity of the core) within the containment when the vessel head is unbolted or removed.

m. RATED POWER

RATED POWER is the steady-state reactor core output of 1,673 MWt.

n. REPORTABLE EVENT

A REPORTABLE EVENT is defined as any of those conditions specified in 10 CFR 50.73.

b. Heatup and Cooldown Limit Curves for Normal Operation

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures TS 3.1-1 and TS 3.1-2. Figures TS 3.1-1 and TS 3.1-2 are applicable for the service period of up to 33[1] effective full-power years.

   A. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.

   B. Figures TS 3.1-1 and TS 3.1-2 define limits to assure prevention of non-ductile failure only. For normal operation other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

   C. The isothermal curve in Figure TS 3.1-2 defines limits to assure prevention of non-ductile failure applicable to low temperature overpressurization events only. Application of this curve is limited to evaluation of LTOP events whenever one or more of the RCS cold leg temperatures are less than or equal to the LTOP enabling temperature of 200°F.

2. The secondary side of the steam generator must not be pressurized > 200 psig if the temperature of the steam generator is < 70°F.

3. The pressurizer cooldown and heatup rates shall not exceed 200°F/hr and 100°F/hr, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is > 320°F.

4. The overpressure protection system for low temperature operation shall be operable whenever one or more of the RCS cold leg temperatures are ≤ 200°F, and the reactor vessel head is installed. The system shall be considered operable when at least one of the following conditions is satisfied:

   A. The overpressure relief valve on the Residual Heat Removal System (RHR 33-1) shall have a set pressure of ≤ 500 psig and shall be aligned to the RCS by maintaining valves RHR 1A, 1B, 2A, and 2B open.

      1. With one flow path inoperable, the valves in the parallel flow path shall be verified open with the associated motor breakers for the valves locked in the off position. Restore the inoperable flow path within 5 days or complete depressurization and venting of the RCS through a ≥ 6.4 square inch vent within an additional 8 hours.

      2. With both flow paths or RHR 33-1 inoperable, complete depressurization and venting of the RCS through at least a 6.4 square inch vent pathway within 8 hours.

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[1] The curves are limited to 31.1 EFPY due to changes in vessel fluence associated with operation at uprated power.

Limit curves for normal heatup and cooldown of the primary Reactor Coolant System have been calculated using the methods discussed above and limited application to ASME Boiler and Pressure Vessel Code Case N-588 to the circumferential beltline weld. The derivation of the limit curves is consistent with the NRC Regulatory Standard Review Plan(8) and Footnote.(9)

Transition temperature shifts occurring in the pressure vessel materials due to radiation exposure have been obtained directly from the reactor pressure vessel surveillance program. As presented in WCAP 14279, Revision 1,(10) weld metal Charpy test specimens from Capsule S indicate that the core region weld metal exhibits the largest shift in RT_{NDT} (250°F).

The results of Irradiation Capsules V, R, P, and S analyses are presented in WCAP 8908,(11) WCAP 9878,(12) WCAP-12020,(13) WCAP-14279,(14) and WCAP-14279, Revision 1(10) respectively. Heatup and cooldown limit curves for normal operation of the reactor vessel are presented in Figures TS 3.1-1 and TS 3.1-2 and represent an operational time period of 33[1] effective full-power years.

The isothermal cooldown limit curve (Figure TS 3.1-2) is used for evaluation of low temperature overpressure protection (LTOP) events. This curve is applicable for 33[1] effective full-power years of fluence (through the end of OPERATING cycle 33[1]). If a low temperature overpressure event occurred, the RCS pressure transient would be evaluated to the limits of this figure to verify the integrity of the reactor vessel. If these limits are not exceeded, vessel integrity is assured and a TS violation has not occurred.

<u>Note</u>:

[1] The curves are limited to 31.1 EFPY due to changes in vessel fluence associated with operation at uprated power.

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(8) "Fracture Toughness Requirements," Branch Technical Position MTEB 5-2, Chapter 5.3.2 in Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, NUREG-0800, 1981.

(9) 1989 ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, Appendix G, "Fracture Toughness Criteria for Protection Against Failure."

(10) C. Kim, et al., "Evaluation of Capsule S from the Kewaunee and Capsule A35 from the Maine Yankee Nuclear Power Reactor Vessel Radiation Surveillance Programs," WCAP-14279, Revision 1, September 1998.

(11) S.E. Yanichko, S. L. Anderson, and K. V. Scott, "Analysis of Capsule V from the Wisconsin Public Service Corporation Kewaunee Nuclear Plant Reactor Vessel Radiation Surveillance Program," WCAP 8908, January 1977.

(12) S.E. Yanichko, et al., "Analysis of Capsule R from the Wisconsin Public Service Corporation Kewaunee Nuclear Plant Reactor Vessel Radiation Surveillance Program," WCAP 9878, March 1981.

(13) S.E. Yanichko, et al., "Analysis of Capsule P from the Wisconsin Public Service Corporation Kewaunee Nuclear Power Plant Reactor Vessel Radiation Surveillance Program," WCAP-12020, November 1988.

(14) E. Terek, et al., "Analysis of Capsule S from the Wisconsin Public Service Corporation Kewaunee Nuclear Power Plant Reactor Vessel Radiation Surveillance Program," WCAP-14279, March 1995.

Pressurizer Limits (TS 3.1.b.3)

Although the pressurizer operates at temperature ranges above those for which there is reason for concern about brittle fracture, OPERATING limits are provided to ensure compatibility of operation with the fatigue analysis performed in accordance with Code requirements. In-plant testing and calculations have shown that a pressurizer heatup rate of 100°F/hr cannot be achieved with the installed equipment.

Low Temperature Overpressure Protection (TS 3.1.b.4)

The Low Temperature Overpressure Protection System must be OPERABLE during startup and shutdown conditions below the enable temperature (i.e., low temperature) as defined in Branch Technical Position RSB 5-2 as modified by ASME Boiler and Pressure Vessel Code Case N–514. Based on the Kewaunee Appendix G LTOP protection pressure-temperature limits calculated through 33[1] effective full-power years, the LTOP System must be OPERABLE whenever one or more of the RCS cold leg temperatures are ≤ 200°F and the head is on the reactor vessel. The LTOP system is considered OPERABLE when all four valves on the RHR suction piping (valves RHR-1A, 1B, 2A, 2B) are open and valve RHR-33-1, the LTOP valve, is able to relieve RCS overpressure events without violating Figure TS 3.1-2.

The set pressure specified in TS 3.1.b.4 includes consideration for the opening pressure tolerance of ± 3% (± 15 psig) as defined in ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NC: Class 2 Components for Safety Relief Valves. The analysis of pressure transient conditions has demonstrated acceptable relieving capability at the upper tolerance limit of 515 psig.

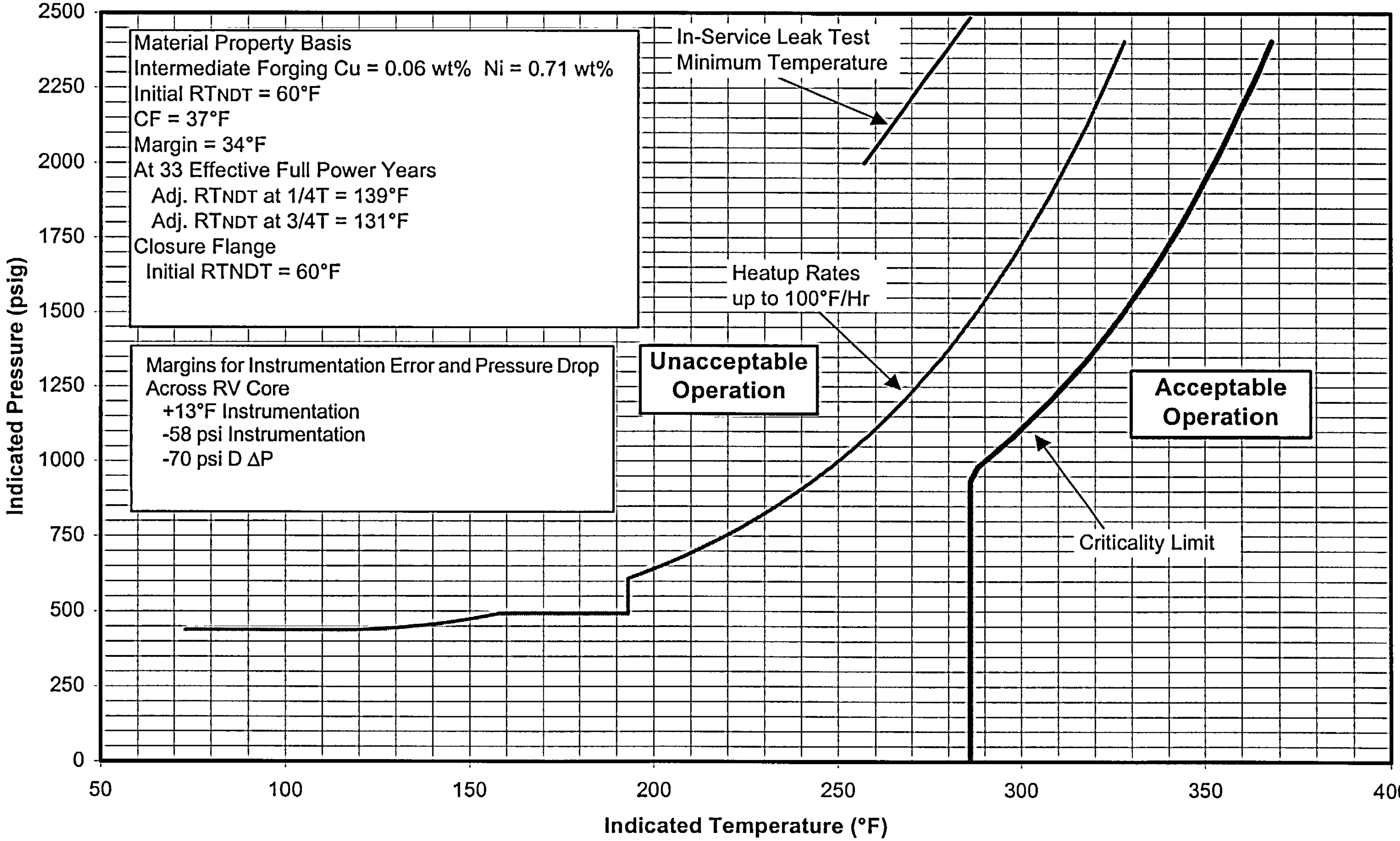
If one train of RHR suction piping to RHR 33-1 is isolated, then the valves and valve breakers in the other train shall be verified open, and the isolated flowpath must be restored within five days. If the isolated flowpath cannot be restored within five days, then the RCS must be depressurized and vented through at least a 6.4 square inch vent within an additional eight hours.

If both trains of RHR suction are isolated or valve RHR 33-1 is inoperable, then the system can still be considered OPERABLE if an alternate vent path is provided which has the same or greater effective flow cross section as the LTOP safety valve (≥ 6.4 square inches). If vent path is provided by physical openings in the RCS pressure boundary (e.g., removal of pressurizer safety valves or steam generator manways),then the vent path is considered secured in the open position.

Note

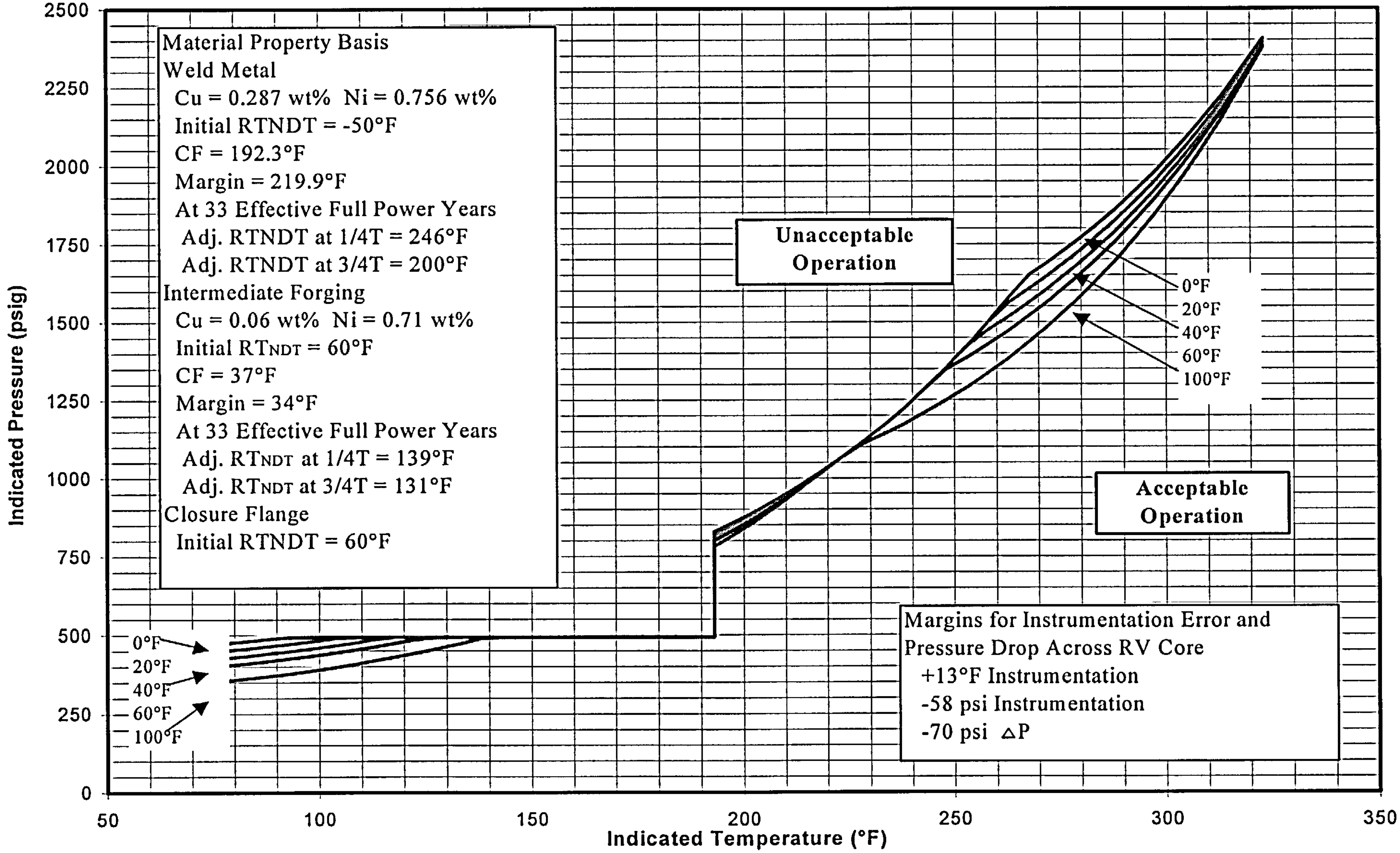
[1] The curves are limited to 31.1 EFPY due to changes in vessel fluence associated with operation at uprated power.

FIGURE TS 3.1-1
KEWAUNEE UNIT NO. 1 HEATUP LIMITATION CURVES
APPLICABLE FOR PERIODS UP TO 33[1] EFFECTIVE FULL-POWER YEARS



NOTE:
[1]The curves are limited to 31.1 EFPY due to changes in vessel fluence associated with operation at uprated power.

FIGURE TS 3.1-2
KEWAUNEE UNIT NO. 1 COOLDOWN LIMITATION CURVES
APPLICABLE FOR PERIODS UP TO 33 [1] EFFECTIVE FULL-POWER YEARS



[1] The curves are limited to 31.1 EFPY
due to changes in vessel fluence associated with operation at uprated powe.

3. Monthly OPERATING Report

Routine reports of OPERATING statistics and shutdown experience shall be submitted on a monthly basis to the Document Control Desk, U.S. Nuclear Regulatory Commission, Washington, D.C., 20555, with a copy to the appropriate Regional Office, to be submitted by the fifteenth of each month following the calendar month covered by the report.

4. Core Operating Limits Report (COLR)

A. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

(1)	TS 2.1	Reactor Core Safety Limit
(2)	TS 2.3.a.3.A	Overtemperature ΔT Setpoint
(3)	TS 2.3.a.3.B	Overpower ΔT Setpoint
(4)	TS 3.1.f.3	Moderator Temperature Coefficient (MTC)
(5)	TS 3.1.f.4	Moderator Temperature Coefficient (MTC)
(6)	TS 3.8.a.5	Refueling Boron Concentration
(7)	TS 3.10.a	Shutdown Margin
(8)	TS 3.10.b.1.A	$F_Q^N(Z)$ Limits
(9)	TS 3.10.b.1.B	$F_{\Delta H}^N$ Limits
(10)	TS 3.10.b.4	$F_Q^{EQ}(Z)$ Limits
(11)	TS 3.10.b.5.C.i	$F_Q^{EQ}(Z)$ penalty
(12)	TS 3.10.b.9	Axial Flux Difference Target Band
(13)	TS 3.10.b.11.A	Axial Flux Difference Envelope
(14)	TS 3.10.d.1	Shutdown Bank Insertion Limits
(15)	TS 3.10.d.2	Control Bank Insertion Limits
(16)	TS 3.10.k	Core Average Temperature
(17)	TS 3.10.l	Reactor Coolant System Pressure
(18)	TS 3.10.m.1	Reactor Coolant Flow

B. "The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC> When an initial assumed power level of 102% of the original rated power is specified in a previously approved method, 100.6% of uprated power may be used only when the main feedwater flow measurement (used as the input for reactor thermal output) is provided by the Crossflow untrasonic flow measurement system (Crossflow system) as described in report (15) listed below. When main feedwater flow measurements from the Crossflow System are unavailable, a power measurement uncertainty consistent with the instrumentation used shall be applied.

"Future revisions of approved analytical methods listed in this Technical Specification that currently reference the original Appendix K uncertainty of 102% of the original rated power should include the condition given above allowing use of 100.6% of uprated power in the safety analysis methodology when the Crossflow system is used for main feedwater flow measurement.

"The approved analytical methods are described in the following documents.

(1) SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION ON "QUALIFICATIONS OF REACTOR PHYSICS METHODS FOR APPLICATION TO KEWAUNEE" REPORT, dated August 21, 1979, report date September 29, 1978

(2) KEWAUNEE NUCLEAR POWER PLANT – REVIEW FOR KEWAUNEE RELOAD SAFETY EVALUATION METHODS TOPICAL REPORT WPSRSEM-NP, REVISION 3 (TAC NO MB0306) dated September 10, 2001.

(3) Nissley, M.E. et, al., "Westinghouse Large-Break LOCA Best-Estimate Methodology," WCAP-10924-P-A, Volume 1, Revision 1, Addendum 4, March 1991, Volume 1: Model Description and Validation; Addendum 4: Model Revisions.

(4) N. Lee et al., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A (Proprietary) and WCAP-10081-NP-A (Non-Proprietary), dated August 1985.

(5) C.M. Thompson, et al., "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condesation Model," WCAP-10054-P-A, Addendum 2, Revision 1 (Proprietary) and WCAP-10081-NP (Non-Proprietary), dated July 1997.

(6) XN-NF-82-06 (P)(A) Revision 1 and Supplements 2, 4, and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup, Exxon Nuclear Company, dated October 1986.

(7) ANF-88-133 (P)(A) and Supplement 1, "Qualification of Advanced Nuclear Fuels' PWR Design Meghodology for Rod Burnups of 62 GWd/MTU," Advanced Nuclear Fuels Corporation, dated December 1991.

(8) EMF-92-116 (P)(A) Revision 0, "Generic Mechanical Design Criteria for PWR Fuel Designs," Siemens Power Corporation, dated February 1999.

(9) XN-NF-77-57, Exxon Nuclear Power Distribution Control for Pressurized Water Reactors, Phase II, dated January 1978, and Supplement 2, dated October 1981.

(10) WCAP-10216-P-A, Rev. 1A, "Relaxation of Constant Axial Offset Control FQ Surveillance Technical Specification," February 1994 (W Proprietary).

(11) WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary)

(12) WCAP-8745-P-A, Design Bases for the Thermal Overtemperature ΔT and Thermal Overpower ΔT trip functions, September 1986.

(13) WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES-TOPICAL REPORT," September 1974 (Westinghouse Proprietary).

(14) WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," April 1995 (Westinghouse Proprietary).

(15) CENP-397-P-A, "Improved Flow Measurement Accuracy Using Cross Flow Ultrasonic Flow Measurement Technology, "Rev. 1, May 2000.

C. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

D. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

b. Unique Reporting Requirements

1. Annual Radiological Environmental Monitoring Report

   A. Routine Radiological Environmental Monitoring Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the OFF-SITE DOSE CALCULATION MANUAL (ODCM) and Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

2. Radioactive Effluent Release Report

   Routine Radioactive Effluent Release Reports covering the operation of the unit for the previous calendar year shall be submitted by May 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and the PCP, and in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

3. Special Reports

   A. Special reports may be required covering inspections, test and maintenance activities. These special reports are determined on an individual basis for each unit and their preparation and submittal are designated in the Technical Specifications.

      (1) Special reports shall be submitted to the Director of the NRC Regional Office listed in Appendix D, 10 CFR Part 20, with a copy to the Director, Office of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555 within the time period specified for each report.

ATTACHMENT 12

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

Dated

January 13, 2003

License Amendment Request 193

List of Regulatory Commitments

LIST OF REGULATORY COMMITMENTS

The following table identifies those actions committed to by NMC in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

COMMITMENT	Due Date/Event
1. KNPP will complete revisions to affected documents (i.e., procedures) and provide appropriate training to the necessary plant staff for changes associated with the installation of the Crossflow UFMD and the implementation of the new rated power (Attachment 2, Sections I.1.D.1 and I.1.F.i).	1. Prior to MUR power uprate implementation.
2. The KNPP will ensure the plant specific analysis has been completed and that the plant specific uncertainties are equal to or less than those provided to Westinghouse for the calculation of the power measurement uncertainty (Attachment 2, Section I.1.C).	2. Prior to MUR power uprate implementation.
3. KNPP will complete revisions to affected operations procedures and provide appropriate training to operations for the implementation of the new rated power and the administrative restrictions for inoperable Crossflow UFMDs (Attachment 2, Sections I.1.H, VII.2.A, VII.2.D, VII.3).	3. Prior to MUR power uprate implementation.
4. The KNPP EQ Plan will be updated to include the new containment exclusion areas for the pressurizer, steam generator, and reactor coolant pump vaults (Attachment 2, Section III.3)	4. Prior to MUR power uprate implementation.
5. A corrective action request has been initiated to investigate the Reserve Auxiliary Transformer procedural limit. This will be completed prior to the MUR power uprate implementation (Attachment 2, Table V-1, "Electrical Equipment Information").	5. Prior to MUR power uprate implementation.
6. Modifications associated with the MUR power uprate will be completed prior to implementation. This includes the installation of the Crossflow UFMDs and implementation of the PPCS and control room alarm functions (Attachment 2, Section VII.3).	6. Prior to MUR power uprate implementation.
7. Rescaling and setting changes of the protection system will be completed as necessary (Attachment 2, Section VIII).	7. Prior to MUR power uprate implementation.