5.3 Fuel Assemblies

5.3.1 Introduction and Background

The structural integrity of the Westinghouse 422V+ fuel assembly (14x14 VANTAGE + with PERFORMANCE+ features) has been evaluated in support of Fuel Uprade and Power Uprate Program as documented in Section 2.5 of the *Reload Safety Transition Report* (Reference 1). The RTSR analyses and evaluations have been performed to support a 7.4-percent power uprate and the transition to 422V+ fuel. Furthermore, the structural evaluation of the Westinghouse 422 V+ fuel assembly was performed for the KNPP for both homogeneous and heterogeneous cores with co-resident Framatome ANP high-thermal performance heavy fuel.

5.3.2 Methodology and Summary

The evaluation of the 422V+ fuel assembly in accordance with NRC requirements as given in the *Standard Review Plant* (SRP) 4.2, Appendix A (Reference 2), shows that the 422V+ fuel is structurally acceptable for the Kewaunee reactors.

The lateral effects of loss-of-coolant accident (LOCA) auxiliary line breaks (accumulator lines) and two safe shutdown earthquake (SSE) (SSE Case A and Case B) seismic accidents were considered for the Kewaunee Uprate Program. With the appropriate analysis parameters, such as grid impact stiffness and damping, number of fuel assemblies in a planar array, and established gap clearance, the reactor core models were used for analyzing lateral transient loading. The detail model of the 422V+ fuel assembly was used for analyzing the vertical impact loading and the maximum lateral deflection.

The fuel assembly stress structural analysis for the Westinghouse fuel was based on a conservative vertical load and an analytically generated fuel assembly lateral deflection of 1.0 inch at grid 4, which are in turn based on combined seismic and LOCA load analyses. The results of the analysis indicated that the Westinghouse fuel assembly design satisfies the design requirements for the fuel rod and guide thimbles by a factor of more than 2.

A comparison of the Framatome fuel to the Westinghouse fuel indicates that the designs are materially and structurally very similar with only minor differences in the diametrical dimensions of the thimble tubes and fuel rods. A comparison of the major material and structural characteristics for the two fuel assembly designs is given in Table 5.3-1.

The conservative vertical loading conditions assumed for the Westinghouse fuel are applicable and conservative for the Framatome fuel as well. The analytically generated lateral displacement of grid 4 for the Framatome design is 1.1 inches, only 10 percent larger than the comparable lateral displacement for the Westinghouse design. Due to the large margin (more than a factor of 2) to the design criteria limits for the Westinghouse fuel, a lateral deflection of 1.1 inches at grid 4 in the Westinghouse fuel could be accommodated with significant margin remaining. Since the Framatome fuel is materially and structurally very similar to the Westinghouse fuel, the Framatome fuel is also expected to meet the fuel assembly stress design criteria with sufficient margin. Furthermore, the effect of the mixed core and uprate conditions on the load and deflection inputs to the fuel assembly stress structural analysis are minimal, so that the margin in the existing analyses is not expected to be significantly eroded.

Based on analysis and engineering evaluation, the mechanical and structural fuel assembly design criteria in mixed core and homogeneous configurations are satisfied for the Westinghouse 422 V+ fuel design and will, with a high degree of confidence, be satisfied for the Framatome fuel design.[1] The basic acceptance criteria for combining SSE and LOCA loading:

- Fuel rod fragmentation does not occur.
- Control rod insertability is maintained.
- A coolable core geometry is maintained.

The grid impact loads of the 422V+ homogenous core evaluated for the LOCA and seismic lateral loading and combined by the square root of the sum of squares (SRSS) method identified in SRP 4.2 are less than the allowable limit. Thus, the core coolable geometry is maintained.

From the two limiting transition cores, the grid impact results of the combined seismic and LOCA analyses indicate that the maximum impact forces for the 422V+ assembly design, using the two-directional grid characteristics, are less than the respective allowable grid strengths (except 13 fuel assemblies row in mixed Condition I). Based on the results of the combined SSE and LOCA loads and the additional core coolable geometry assessment, the 422V+ fuel assembly is structurally acceptable for the Kewaunee Nuclear Power Plant (KNPP) for both homogenous

[1] NMC assumes responsibility for this position.

core and transition cores. The maximum impact forces of the Framatome fuel in the limiting transition cores are also less than the allowable strength.

The stresses in the 422V+ fuel assembly components resulting from seismic and LOCA vertical impact load and the induced maximum deflection are within acceptable limits. The results indicate that there is adequate margin for fuel rods and thimble tubes, eliminating the possibility of fuel rod fragmentation. The reactor can be safely shutdown under the combined faulted condition loads.

5.3.3 Conclusions

The results of the RTSR analysis are applicable to Kewaunee for a 7.4-percent power uprate and for the transition to 422V+ fuel. The results also show adequate grid load margin and that the core coolable geometry and control rod insertion requirements are met. In terms of the structural analysis; comparisons of the Framatome and Westinghouse fuel indicate that similar designs, combined with conservative analytical assumptions, result in sufficient margin to meet the design criteria.

5.3.4 References

1. KEW-RTSR-02-021, *Kewaunee Nuclear Power Plant, RTSR Program*, RTSR, R. H. Owoc, July 19, 2002, attaches Sisk, R. B., *Reload Transition Safety Report for the Kewaunee Nuclear Power Plant*, July 2002 (NRC Submittal via NMC Letter NRC-02-067, *License Amendment Request 187 to the Kewaunee Nuclear Power Plant Technical Specifications. Conforming Technical Specification Changes for Use of Westinghouse VANTAGE+ Fuel*_(Docket 50-305), July 26, 2002.

2. *Standard Review Plan* (SRP) 4.2, Appendix A.

Table 5.3-1 Westinghouse and Framatone Fuel Structural Characteristic Comparisons		
	14 x 14 Westinghouse 422V+	**Framatone Siemens**
FUEL ASSEMBLY		
Rod Array in Assembly	14 x 14	14 x 14
Rods per Assembly	179	179
Assembly Pitch, in	7.803	7.803
Overall Assembly Envelope, in	7.761	7.761
Overall Assembly Height, in	159.775	159.710
GUIDE THIMBLE/INSTRUMENT TUBE		
Number of Guide Thimbles	16	16
Guide Thimble Material	ZIRLO™	Zircaloy-4
GT O.D. above Dashpot, in	0.526	0.541
GT I.D. above Dashpot, in	0.492	0.507
GT Thickness above Dashpot, in	0.017 nom. or 0.0147 min	0.017
GT O.D. in Dashpot, in	0.4805	0.481
GT I.D. in Dashpot, in	0.4465	0.447
GT Thickness in Dashpot, in	0.017 nom. or 0.0147 min	0.017
Number of Instrument Tubes	1	1
Instrument Tube Material	ZIRLO™	Zircaloy-4
Instrument Tube O.D., in	0.422	0.424
Instrument Tube I.D., in	0.3734	0.374
GRIDS		
Total Number of Grids	7	7
Number of Top/Bottom Grids	1/1	1/1
FUEL ROD		
Cladding Material	ZIRLO™	Zircaloy-4
Cladding O.D., in	0.422	0.424
Cladding I.D., in	0.3734	0.374
Cladding Thickness, in	0.0243	0.025

5.4 Control Rod Drive Mechanisms

5.4.1 Introduction and Background

This section addresses the ASME Code of record structural considerations for the pressure boundary components of the Westinghouse full-length L-106A control rod drive mechanisms (CRDMs). The CRDMs are evaluated for the Kewaunee Power Uprating Program Performance Capability Working Group (PCWG) parameters (Section 2) and the Nuclear Steam Supply System (NSSS) design transients (Section 3.1), which assumed a 7.4-percent increase in core power.

5.4.2 Input Parameters and Assumptions

The Model L-106A CRDMs were originally designed and analyzed to meet the ASME Code 1965 Edition through the Summer 1966 Addenda or later (Reference 1). The original analyses were contained in Reference 2. A later evaluation of the Model L-106A CRDMs was conducted for the Replacement Steam Generator (RSG) Project for the Kewaunee plant.

The current Kewaunee Power Uprating Program modifies the PCWG parameters (see Section 2) and the NSSS design transients (see Section 3.1) that were considered in these previous evaluations. The seismic loading has not been changed for the Kewaunee Power Uprating Program.

The Kewaunee CRDMs are of the hot head type, defined by the vessel outlet reactor coolant temperature on the PCWG parameters, and must be analyzed for the NSSS design transients defined for the hot leg.

The differences associated with the uprating requirements are discussed in subsection 5.4.3.

5.4.3 Description of Analysis

5.4.3.1 Operating Pressure and Temperature

There are no changes from the current reactor coolant pressure of 2250 psia for any of the uprating cases from the Power Uprating Program PCWG parameters for Kewaunee. The hot-leg temperature (T_{hot}) defined by the vessel outlet temperature on the PCWG parameters for the Kewaunee Power Uprating Program is a maximum of 606.8°F, which is the same as the temperature defined for the RSG Project. Since none of the temperatures exceeds the previously considered temperature, and the pressure does not change, the Power Uprating Program PCWG parameters are bounded by the original and RSG analyses.

Table 5.4-1 summarizes the hot leg PCWG parameters. From Table 5.4-1, the present (RSG) conditions provide a RCS T_{hot} range of 586.3° to 606.8°F, compared to 590.8° to 606.8°F for the Power Uprating Program. Therefore, the present operation range bounds the range of T_{hot} for the Power Uprating Program.

5.4.3.2 Transient Discussion

The only hot-leg transients that have been modified to become more severe for the uprating are the large-step decrease and the reactor trip from full power. For the large-step decrease, the change in temperature for the low-temperature operating condition becomes -78.5°F. For the RSG evaluation, the controlling temperature change for this transient was -77°F for the high-temperature operating condition. For the reactor trip transient, the temperature change for the low temperature operating case becomes -68.8°F. For the RSG evaluation, as well as the low-temperature operating condition, the controlling temperature change for this transient was -64.8°F. These changes are addressed in subsection 5.4.5 for the Power Uprating Program.

There are no changes in the pressure transients associated with these system transients.

5.4.4 Acceptance Criteria and Results

The acceptance criteria for the ASME Code structural analysis of the CRDM pressure boundary are that the analyzed stresses do not exceed the stress allowables of the ASME Code and that the cumulative usage factors from the Code fatigue analysis remain less than 1.0.

When the RSG evaluation was performed, the stresses and the cumulative usage factors were increased where changes to the design transients indicated that an increase was necessary. However, where changes to the design transients would have allowed a decrease in stresses or cumulative usage factors, no decrease was calculated, and no credit was taken for such a decrease. Therefore, if the NSSS design transients are shown to be bounded by those considered for either the RSG Project or the original design, then the stresses and cumulative usage factors calculated for the CRDMs for the RSG Project remain bounding and applicable for the Power Uprating Program.

5.4.5 Results

The only difference from the previous (RSG) evaluation and the current Power Uprating Program, as discussed in subsection 5.4.3.2, is the modification of the large-step decrease and the reactor-trip transients.

The operating temperature and pressure discussion presented above showed that the operating pressures and temperatures were bounded by those considered for the RSG Project.

For the original evaluation of the CRDMs (Reference 2), the large-step decrease had a temperature change of -84°F, and the reactor trip had a temperature change of -78°F (-200°F for the reactor trip – rod drop). These values are more severe than the -78.5° and -68.8°F temperature changes defined for these transients for the current Power Uprate Program. As discussed in subsection 5.4.4, the stresses and cumulative usage factors calculated for the RSG Project bound both the RSG Project and the original analysis conditions. Power uprate operating parameters and design transients are bounded by the operating parameters and design transients used for the current CRDM qualification, therefore, CRDMs are acceptable for power uprate operation.

A summary of the results of the analysis performed for the RSG Program is presented in Tables 5.4-2 and 5.4-3. The highest recalculated stresses, as compared to the associated allowables, are presented in Table 5.4-2 for the upper, middle, and lower joints of the CRDM pressure boundary. The cumulative usage factors that were recalculated for the RSG Program are given in Table 5.4-3. It is noted that the highest cumulative usage factor, [ ]a,c at the upper joint canopy, was calculated in a conservative manner where the applied transients were grouped for analysis and the allowable number of cycles considered for each group was based on the most severe transient in the group.

5.4.6 Conclusions

The Power Uprate Program PCWG parameters and NSSS design transients have been shown to be bounded by the parameters and transients considered for either the RSG Project or the original design analysis. Therefore, the conclusions of the RSG Project evaluation are still valid and applicable to the Power Uprating Program. The CRDMs are acceptable from a structural standpoint. The CRDM pressure boundary parts still satisfy the ASME Code of record.

Therefore, the evaluation results for the Power Uprate Program are consistent with, and continue to comply with, the current licensing basis/acceptance requirements for Kewaunee.

5.4.7 References

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

2. Engineering Memorandum No. 4531, *Stress and Thermal Report of Type L106A and L106B CRDM, S. O. M312*, Rev. 2, S. K. Ganguly, J. R. Raymond, and A. E. Reed, Westinghouse Electro-Mechanical Division, April 12, 1976.

Table 5.4-1 **PCWG Conditions Used to Bracket All Operating Conditions for Kewaunee 7.4-Percent Power Uprating**				
	Present		**Uprating**	
Parameter	**High T_{avg}**	**Low T_{avg}**	**High T_{avg}**	**Low T_{avg}**
T_{hot}	606.8°F	586.3°F	606.8°F	590.8°F

Table 5.4-2 **Highest Stresses, Compared to Allowables, for CRDM Joints, as Recalculated for the RSG Project and Remaining Applicable for Kewaunee 7.4-Percent Power Uprating**		
	Stress (psi)	
CRDM Joint and Component	**Value Calculated for the RSG Project**	**Allowable Value**
Upper Joint Threaded Area	[ ]^{a,c}	20,620
Middle Joint Canopy	[ ]^{a,c}	45,900
Lower Joint Canopy	[ ]^{a,c}	45,900

Bracketed [ ]^{a,c} information designates data that is Westinghouse Proprietary, as dicussed in Section 1.7 of this report.

Table 5.4-3

Cumulative Usage Factors for CRDM Joints, as Recalculated for the RSG Project and Remaining Applicable for Kewaunee 7.4-Percent Power Uprating

CRDM Joint and Component	Cumulative Usage Factor: Value Calculated for the RSG Project	Cumulative Usage Factor: Allowable Value
Upper Joint Canopy	[ ]a,c	1.00
Upper Joint Canopy Weld	[ ]a,c	1.00
Upper Joint Threaded Area	[ ]a,c	1.00
Middle Joint Canopy Weld	[ ]a,c	1.00
Lower Joint Canopy Weld	[ ]a,c	1.00

Bracketed [ ]a,c information designates data that is Westinghouse Proprietary, as dicussed in Section 1.7 of this report.

5.5 Reactor Coolant Loop Piping and Supports

5.5.1 Reactor Coolant Loop Piping

5.5.1.1 Introduction and Background

The parameters associated with the 7.4-percent Power Uprate Project, which also support the *KNPP Reload Transition Safety Report* (RTSR) (Reference 1), were reviewed for impact on the existing design basis replacement steam generator (RSG) reactor coolant loop (RCL) analysis for the following components:

- RCL piping stresses and displacements
- Primary equipment nozzle loads
- Primary equipment support loads
- Pressurizer surge line piping stresses and displacements including the effects of thermal stratification
- Impact on RCL branch nozzle loads
- Impact on Class 1 auxiliary piping systems

5.5.1.2 Inputs and Assumptions

The following three basic sets of input parameters were considered in the evaluation:

- Nuclear Steam Supply System (NSSS) Performance Capability Working Group (PCWG) design parameters (Table 2.1-1)
- NSSS design transients (Section 3 of this report)
- LOCA hydraulic forcing functions loads (Section 6.6) and associated reactor pressure vessel (RPV) motions

The acceptance criteria for the RCL analysis (USAS B31.1 Power Piping Code, 1967 Edition, Reference 2) are as specified in Reference 3.

The acceptance criteria for the pressurizer surge line thermal stratification analysis (*American Society of Mechanical Engineers [ASME] Boiler & Pressure Vessel [B&PV]* Code Section III, Subsection NB, 1986 Edition, Reference 4), are as specified in WCAP-12841 (Reference 5).

The acceptance criteria for the primary equipment supports (*American Institute of Steel Construction [AISC] Specification for the Design, Fabrication, & Erection of Structural Steel Buildings*, 1969, Reference 6), are as specified in WCAP-7840 (Reference 3).

The parameters associated with the 7.4-percent Power Uprate Program were reviewed for impact on the existing RCL piping and subsequent impact to the RCL branch nozzles and the Class 1 auxiliary lines attached to the RCL. The conclusions of this review are summarized below.

Nuclear Steam Supply System Performance Capability Working Group Design Parameters

The PCWG design parameters as identified in Table 2.1-1 of this report were compared with the design basis thermal analysis for the RCL. The changes in these parameters for the 7.4-percent Uprate Program, as compared to the corresponding PCWG parameters for the RSG Program, are very small (see Table 5.5-1). The RCL is evaluated for two temperature cases as identified in Table 5.5-1. These two cases bound the 100-percent power operating window as defined by the uprated parameters. Therefore, the impact on the RCL analysis is determined to be insignificant. It is also noted that because the uprated PCWG parameters are bounded by the RSG parameters (that is, maximum and minimum temperatures), there is no adverse impact to the primary equipment support thermal gaps and shims. The shimming of the primary equipment supports as performed for the RSG Program is still applicable. Additionally, the current design basis analysis of the pressurizer surge line envelopes the temperatures identified in Table 2.1-1. Therefore, the design basis results summarized in Reference 4 for the pressurizer surge line remain applicable.

Nuclear Steam Supply System Design Transients

The impact on design transients due to the changes in full-power operating temperatures for the 7.4-percent power uprate is addressed in Section 3. As per Section 3, for all primary side components seeing the hot-leg temperature (T_{hot}, based on the high T_{avg} condition) and cold-leg temperature (T_{cold}, based on the low T_{avg} condition), the limiting cases remain unchanged from the RSG Program transients. Additionally, per Reference 5, the design criteria for the RCL piping is USAS B31.1 Power Piping Code, 1967 Edition; thus, no fatigue analysis is required for the RCL.

For the pressurizer surge line, the impact of the design transients with respect to the thermal stratification and fatigue analysis is controlled by the ΔT between the pressurizer temperature and the hot-leg temperature. As per Section 3, the RSG Program had the larger temperature window for design purposes and the limiting full-power T_{hot} and T_{cold} values for the uprating are bounded by the ones used in the RSG Program. Therefore, the 7.4-percent power uprate has no adverse impact on either the thermal stratification or the fatigue analysis for the pressurizer surge line, and the results in Reference 4 remain valid.

Loss-of-Coolant-Accident Hydraulic Forcing Functions Loads and Associated Reactor Pressure Vessel Motions

The impact on the LOCA hydraulic forcing functions (HFFs) due to the 7.4-percent Power Uprate Program is addressed in Section 6.6 of this report. LBB is applicable for the RCL main loop piping and the pressurizer surge line (refer to subsection 5.5.2). However, it is noted that the RCL is evaluated for LOCA using HFFs generated for the Uprate Program conservatively, based on breaks at the 10-inch surge line nozzle on the hot leg, and at the 12-inch accumulator line nozzle on the cold leg. RPV motions corresponding to the surge line break and accumulator line break are also included.

Additionally, the RCL was also evaluated for secondary side breaks at the main steam line and feedwater line terminal end nozzle locations at the steam generator. Based on the PCWG design parameters identified in Table 2.1-1, the secondary side breaks at the main steam line and feedwater line terminal end nozzle locations remain bounded by the evaluation performed for the RSG Program due to the reduction in steam generator operating pressure.

5.5.1.3 Analysis Methods

As previously defined in Reference 3, the system analysis of the RCL piping was performed using program WESTDYN for all applicable deadweight, thermal expansion, seismic, and LOCA cases. The analysis of the RCL primary equipment supports was performed using program STAAD-III.

5.5.1.4 Reactor Coolant Loop Piping Analysis and Results

The evaluation of the RCL includes performing deadweight, thermal expansion, seismic, and LOCA analyses.

The deadweight analysis is not impacted by the 7.4-percent Power Uprate Program. Therefore the evaluation as performed for the RSG Program is still applicable.

The thermal analysis considered the range of operating temperatures for 100-percent power as defined by the uprated NSSS parameters identified in Tables 2.1-1 and 5.5-1 of this report. The RCL primary equipment supports were shimmed to accommodate the range of operating temperatures as specified by the RSG Program. Because these temperature ranges bound corresponding temperature ranges for the 7.4-percent Power Uprate Program, the shimming performed for the RSG Program is not impacted by the Uprate Program.

The seismic analysis methods and input response spectra are not impacted by the 7.4-percent power uprate parameters. Seismic analysis for the RSG Program was performed for operating basis earthquake (OBE) and a factor of 2.0 applied to obtain safe shutdown earthquake (SSE) conditions. The seismic analyses performed considered multiple cases based on various primary equipment support activity, and accounted for the range of operating temperatures as defined by the uprated PCWG parameters.

The LOCA analysis for the RCL is performed considering time history hydraulic forces distributed throughout the RCL system. The analysis is performed for the breaks at the auxiliary nozzles for the 8-inch residual heat removal (RHR) line on the hot leg, and the 12-inch accumulator line on the cold leg. As previously noted, KNPP has been licensed for LBB on the main loop piping and the pressurizer surge line. However, the HFFs utilized in the LOCA evaluations for the 8-inch RHR line are conservatively based on a 10-inch surge line break for the 7.4-percent Power Uprate Program. The LOCA analyses performed considered multiple cases based on various primary equipment support activity, and accounted for the range of operating temperatures as defined by the uprated PCWG parameters.

Based on the evaluations performed for the uprated NSSS PCWG design parameters, NSSS design transients, LOCA HFFs and associated RPV motions, it is concluded that there is no adverse effect on the current design basis RSG RCL analyses and the current design basis remains acceptable for the 7.4-percent Power Uprate Program.

The maximum RCL piping stresses for the RCL piping and the corresponding code-allowable stress values are presented in Table 5.5-2. The stresses are combined in accordance with the methods and code criteria as described in Reference 3.

The primary equipment nozzle loads are compared to the allowables as defined in the equipment design specifications and the loads previously evaluated for the RSG Program. The nozzle loads are acceptable and the 7.4-percent Power Uprate Program has no adverse impact on analysis results.

The applicable RCL piping loads resulting from the range of operating temperatures, as defined by the uprated NSSS parameters and bound by the RSG parameters, were provided for evaluation and confirmation of LBB (see subsection 5.5.2).

The impact on RCL piping displacements at the RCL branch nozzles and corresponding Class 1 auxiliary piping systems is insignificant due to the uprated parameters and revised LOCA hydraulics. Therefore, the subsequent impact on the RCL branch nozzles and the corresponding Class 1 piping systems due to the power uprate, including the Primary Sampling System (PSS), Chemical and Volume Control System (CVCS), Residual Heat Removal System (RHRS), and Safety Injection System (SIS), is considered negligible. The RCL piping and equipment displacements for the RSG Program remain valid.

Additionally, the current design basis pressurizer surge line analysis results including the effects of thermal stratification in Reference 5 are applicable for the 7.4-percent power uprate Program.

5.5.1.5 Primary Equipment Supports Analysis and Results

The primary equipment supports have been evaluated for the revised loads associated with the 7.4-percent Power Uprate Program. The equipment supports are evaluated to meet the stress criteria as defined in Reference 3. The stress limits correspond to the *AISC Specification for the Design, Fabrication, & Erection of Structural Steel Buildings* (Reference 6). Support stresses are evaluated for normal, upset, emergency, and faulted conditions.

The equipment support loads associated with the 7.4-percent power uprate were compared with the loads previously evaluated and qualified for the RSG Program. The stress ratios were adjusted, where necessary, to reflect these new loads due to the power uprate.

Based on the evaluations performed, it is concluded that the primary equipment support stresses remain within the acceptance criteria. Stress ratios are less than 1.0 and are summarized in Table 5.5-3.

In addition, it has been determined that building interface loads do not require further evaluation since these loads are enveloped by previously analyzed loads. Therefore, the loads for the RSG Program remain valid.

Furthermore, since the high and low RCL temperatures associated with the 7.4-percent Power Uprate Program are bounded by the high and low temperatures associated with the RSG, the equipment support thermal gap and shim evaluation performed for the RSG Program remains valid.

5.5.1.6 Conclusions

The parameters associated with the 7.4-percent Power Uprate Program have been evaluated for the following components:

- RCL piping stresses and displacements
- Primary equipment nozzle loads
- Primary equipment support loads
- Pressurizer surge line piping stresses and displacements, including the effects of thermal stratification
- Impact on RCL branch nozzle loads
- Impact on Class 1 auxiliary piping systems

The evaluation indicates that the parameters associated with the 7.4-percent Power Uprate Program have no adverse effect on the analysis of the RCL piping system, including impacts to the primary equipment nozzles and primary equipment supports. RCL piping stresses meet the required stress criteria as summarized in Table 5.5-2. RCL piping loads for LBB evaluation for the 7.4-percent Power Uprate Program were evaluated and found to be acceptable (refer to subsection 5.5.2). RCL primary equipment support loads met the required stress criteria as summarized in Table 5.5-3.

RCL piping displacements at branch nozzles previously provided for the RSG Program remain valid for the power uprate. Therefore, the 7.4- percent power uprate has no subsequent impact to either the RCL branch nozzle loads or the Class 1 auxiliary piping systems that are attached to the RCL.

Additionally, the current design basis analysis results for the pressurizer surge line as documented in Reference 5, including the effects of thermal stratification, are still applicable and remain valid for the 7.4-percent Uprate Program.

5.5.2 Application of Leak-before-Break Methodology

The current structural design basis of KNPP includes the application of LBB methodology to eliminate consideration of the dynamic effects resulting from pipe breaks in the Reactor Coolant System (RCS) primary loop piping and the pressurizer surge line piping. This section describes the analyses and evaluations performed to demonstrate that the elimination of these breaks continues to be justified at the operating conditions associated with the 7.4-percent Power Uprate Program.

5.5.2.1 Reactor Coolant System Primary Loop Piping

5.5.2.1.1 Introduction and Background

Westinghouse performed analyses for the LBB of KNPP primary loop piping in 1987. The results of the analyses were documented in WCAP-11411, Revision 1 (Reference 7) and WCAP-11619 (Reference 8) and approved by the Nuclear Regulatory Commission (NRC) (Reference 9). Westinghouse also performed analyses to support RSG and as well as performed an evaluation for the RSG shim support gap. The results of the RSG analyses were documented in WCAP-15311 (Reference 10).

To demonstrate the elimination of RCS primary loop pipe breaks, the following objectives had to be achieved:

- Demonstrate that margin exists between the "critical" crack size and a postulated crack that yields a detectable leak rate.
- Demonstrate that there is sufficient margin between the leakage through a postulated crack and the leak detection capability.
- Demonstrate margin on the applied load.
- Demonstrate that fatigue crack growth is negligible.

These objectives were met and are documented in References 7, 8, and 10.

To support the 7.4-percent Power Uprate Program at KNPP, the current LBB analyses were updated to address the power uprate conditions. The power uprate evaluation and results are addressed below.

5.5.2.1.2 Input Parameters and Assumptions

The loadings, operating pressure, and temperature parameters for the uprating were used in the evaluation.

The parameters, which are important in the evaluation, are the piping forces, moments, normal operating temperature, and normal operating pressure. These parameters were used in the evaluation. For normal operating temperature and normal operating pressure at the 7.4-percent power uprate conditions, see Section 2 of this report.

5.5.2.1.3 Description of Analyses and Evaluations

The recommendations and criteria proposed in the *Standard Review Plan* (SRP) (Reference 11) are used in this evaluation. The primary loop piping dead weight, normal thermal expansion, and SSE and pressure loads due to the 7.4-percent Power Uprate Program have been used. The normal operating temperature and pressure due to the 7.4-percent power uprate conditions were used in the evaluation. The evaluation showed that all the LBB recommended margins were satisfied for the 7.4-percent power uprate conditions. The margins from Reference 11 are also described below.

5.5.2.1.4 Acceptance Criteria and Results

The LBB acceptance criteria is based on the SRP 3.6.3 (Reference 11). The recommended margins are as follows:

- Margin of 10 on leak rate
- Margin of 2.0 on flaw size
- Margin on loads of 1.0 (Using faulted load combinations by absolute summation method)

The evaluation results showed the following at all the critical locations:

<u>Leak Rate</u> - A margin of 10 exists between the calculated leak rate from the leakage flaw and the leak detection capability of 1 gpm.

<u>Flaw size</u> - A margin of 2.0 or more exits between the critical flaw and the flaw having a leak rate of 10 gpm (the leakage flaw).

<u>Loads</u> - A margin of 1.0 on loads exists.

The evaluation results show that the LBB conclusions provided in References 7, 8, and 10 for KNPP remain unchanged for the 7.4-percent power uprate conditions.

5.5.2.1.5 Conclusions

The LBB acceptance criteria are satisfied for the KNPP primary loop piping at the 7.4-percent power uprate conditions. All the recommended margins are satisfied and the conclusions shown in References 7, 8, and 10 remain valid. It is, therefore, concluded that the dynamic effects of RCS primary loop pipe breaks need not be considered in the structural design basis of KNPP at the 7.4-percent power uprate conditions.

5.5.2.2 Kewaunee Nuclear Power Plant Pressurizer Surge Line Piping

The KNPP pressurizer surge line analysis for the application of LBB was documented in WCAP-12875 (Reference 12) and approved by the NRC (Reference 13). Subsection 5.5.1.4

describes how the current design basis pressurizer surge line analysis results, including the effects of thermal stratification, are applicable to the 7.4-percent Power Uprate Program. Therefore, the conclusions of the previous LBB analysis shown in Reference 12 for KNPP for the pressurizer surge line also remain valid for the 7.4-percent power uprate conditions.

It is, therefore, concluded that the dynamic effects of the pressurizer surge line pipe break need not be considered in the structural design basis of KNPP at the 7.4-percent power uprate conditions.

5.5.2.3 Leak-before-Break Analyses for the Auxiliary Lines

LBB analyses performed for the auxiliary lines are applicable for the 7.4-percent power uprate conditions.

5.5.3 References

1. KEW-RTSR-02-021, *Kewaunee Nuclear Power Plant, RTSR Program*, RTSR, R. H. Owoc, July 19, 2002, attaches Sisk, R. B., *Reload Transition Safety Report for the Kewaunee Nuclear Power Plant*, July 2002 (NRC Submittal via NMC Letter NRC-02-067, *License Amendment Request 187 to the Kewaunee Nuclear Power Plant Technical Specifications. Conforming Technical Specification Changes for Use of Westinghouse VANTAGE+ Fuel* (Docket 50-305), July 26, 2002.

2. *United States of America Standards (USAS)*, B31.1 Power Piping Code, 1967 Edition.

3. WCAP-7840, *Structural Analysis of Reactor Coolant Loop/Support System For NSP (Prairie Island) and WPS (Kewaunee) Nuclear Power Plants*, Report No. SD 103, February 1972.

4. American Society of Mechanical Engineers *Boiler & Pressure Vessel Code*, Section III, Subsection NB, 1986 Edition.

5. WCAP-12841, *Structural Evaluation of the Kewaunee Pressurizer Surge Line, Considering the Effects of Thermal Stratification*, March 1991.

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8. WCAP-11619, *Additional Technical Bases for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Kewaunee*, October 1987.

9. NRC Docket No. 50-305, *Application of Leak-before-Break Technology as a Basis for Kewaunee Nuclear Power Plant Steam Generator Snubber Reduction*, February 16, 1988.

10. WCAP-15311, *Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Kewaunee Nuclear Power Plant after Steam Generator Replacement*, June 2000.

11. *Standard Review Pla*n: Public Comments Solicited, 3.6.3 "Leak-before-Break Evaluation Procedures," Federal Register/Vol. 52, No. 167, notices, pp. 32626 - 32633, Friday, August 28, 1987.

12. WCAP-12875, *Technical Justification for Eliminating Pressurizer Surge Line Rupture as the Structural Design Basis for Kewaunee Nuclear Plant*, June 1991.

13. NRC Docket No. 50-305, *Kewaunee Nuclear Power Plant: Leak-before-Break Evaluation of Pressurizer Surge Line (TAC No. M72140)*, January 3, 1992.

Table 5.5-1

Comparison of PCWG Parameters – 7.4-Percent Power Uprate versus RSG

Thermal Case	**Hot Leg (°F)**	**Crossover Leg (°F)**	**Cold Leg (°F)**
7.4-% Uprate – Cases 1 & 2	590.8	521.6	521.9
7.4-% Uprate – Cases 3 & 4	606.8	538.9	539.2
RSG Cases 1 & 2	586.3	521.6	521.9
RSG Cases 3 & 4	606.8	543.6	543.8
RCL Thermal Analysis			
Thermal Case A (1) (high T_{avg})	612.0	548.0	548.0
Thermal Case B (2) (low T_{avg})	590.8	521.6	521.6

Notes:

1. Case A: represents original design basis thermal, envelopes high T_{avg} thermal case
2. Case B: represents low T_{avg} thermal case

Table 5.5-2

RCL Stress Analysis Summary – 7.4-Percent Power Uprate Program

Stress Condition	Hot Leg Maximum ksi	Hot Leg Allowable ksi	Crossover Leg Maximum ksi	Crossover Leg Allowable ksi	Cold Leg Maximum ksi	Cold Leg Allowable ksi
Equation 9 Design Stress (ksi) (DW, P)	[ ]^{a,c}	14.950	[ ]^{a,c}	14.950	[ ]^{a,c}	14.950
Allowable Stress Limit	(1.0 S)		(1.0 S)		(1.0 S)	
Equation 9 Upset Stress (ksi) (DW, P, OBE)	[ ]^{a,c}	17.940	[ ]^{a,c}	17.940	[ ]^{a,c}	17.940
Allowable Stress Limit	(1.2 S)		(1.2 S)		(1.2 S)	
Equation 9 Faulted Stress (ksi) (DW, P, SSE, LOCA)	[ ]^{a,c}	26.910	[ ]^{a,c}	26.910	[ ]^{a,c}	26.910
Allowable Stress Limit (1)	((1.5 x 1.2S))		((1.5 x 1.2S))		((1.5 x 1.2S))	
Thermal (ksi)	[ ]^{a,c}	25.612	[ ]^{a,c}	25.612	[ ]^{a,c}	25.612
	(1.25 x Sc + 0.25 x Sh)		(1.25 x Sc + 0.25 x Sh)		(1.25 x Sc + 0.25 x Sh)	

Notes:

DW = Deadweight

P = Pressure

1. Faulted stresses conservatively evaluated against emergency limits.

Bracketed [ ]^{a,c} information designates data that is Westinghouse Proprietary, as dicussed in Section 1.7 of this report.

Table 5.5-3

Summary of RSG and RCP Support Member Stress Ratios – 7.4-Percent Power Uprate Program

Support	Support Item	Normal	Upset	Emergency	Faulted
	SGLS Columns	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}
	SGLS Column Foot Bolt Modification	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}
SGLS Vertical	SGLS Column Adapter Modification	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}
	SGLS Bumpers	N/A	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}
	SGLS Bumper Guide Mod.	N/A	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}
SGLS Lateral	SGLS Cross-Compartment Beams	N/A	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}
	SGUS Snubber	N/A	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}
SGUS Lateral	SGUS Bumpers	N/A	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}
SGUS Ring Girder	SGUS Ring Girder	N/A	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}
	SGUS Ring Girder Splice Plate Modification	N/A	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}
RCP Vertical	RCP Columns	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}
RCP Lateral	RCP Tie Rods	N/A	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}
	RCP Tie Rods Bracket Modification	N/A	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}

Notes :

SGLS = steam generator lower support

SGUS = steam generator upper support

All stress ratios are less than 1.0 and are therefore acceptable.

Bracketed [ ]^{a,c} information designates data that is Westinghouse Proprietary, as dicussed in Section 1.7 of this report.

5.6 Reactor Coolant Pumps

The reactor coolant pumps (RCPs) at Kewaunee were evaluated for the power uprating in two separate areas: the structural adequacy of the pumps (subsection 5.6.1 of this report), and the acceptability of the RCP motors (subsection 5.6.2).

5.6.1 Reactor Coolant Pumps (Structural)

5.6.1.1 Introduction and Background

This section addresses the ASME Code structural considerations for the pressure boundary components of the Westinghouse Model 93A RCPs. The Kewaunee RCP equipment specification (Reference 1) requires that the design analysis, materials, welding, inspection, and testing of the pumps meet the requirements of the ASME Code, Section III, 1968 Edition or later (Reference 2). The Kewaunee RCPs predate the inclusion of pumps in the ASME Code, Section III, and are not Code-stamped. As the Code is used for guidance, Code addenda and Editions later than the Code in effect at the time of the order are used for the analyses. The Code editions used in the generic stress reports (References 3 through 6) applicable to the Kewaunee pumps range from the 1968 Edition with Winter 1970 Addenda, to the 1971 Edition with Winter 1972 Addenda.

The RCPs were evaluated for the Kewaunee 7.4-percent Power Uprating Program Performance Capability Working Group (PCWG) parameters (Section 2), and the Nuclear Steam Supply System (NSSS) design transients (Section 3.1), which assumed a 7.4-percent increase in core power.

Reference 6 provides a structural evaluation of the pump non-pressure components. The analyses of Reference 6 are independent of the PCWG parameters and the NSSS design transients. The analyses that are provided in Reference 6 remain valid for the Power Uprating Program conditions.

5.6.1.2 Input Parameters and Assumptions

The Model 93A RCPs were originally designed and analyzed to meet the Kewaunee RCP equipment specification (Reference 1) and the ASME Code (Reference 2). A later evaluation of the RCPs was performed for the Kewaunee Replacement Steam Generator (RSG) Project. The RSG Project evaluation supplements the original evaluation.

The current Kewaunee Power Uprating Program modifies the PCWG parameters (see Section 2 of this report) and the NSSS design transients (see Section 3.1 of this report) that were considered in these previous evaluations. Seismic loadings, nozzle loadings on main and auxiliary nozzles, and auxiliary nozzle transients are either unchanged or remain bounded by the original or the RSG Project basis.

The Kewaunee RCPs are installed in the Reactor Coolant System (RCS) cold leg, between the steam generator outlet and the reactor vessel inlet. The temperatures and pressures used as inputs to the RCP Code structural analysis are those defined for the reactor vessel inlet in the PCWG parameters. The RCPs must be evaluated for the NSSS design transients, as defined for the RCS cold leg by the equipment specification (Reference 1), and updated by the RSG Project and this Power Uprating Program.

5.6.1.3 Description of Analysis

5.6.1.3.1 Operating Temperature and Pressure

From the 7.4-percent Power Uprating Program PCWG parameters for Kewaunee, there are no changes from the current reactor coolant pressure of 2250 psia for any of the uprating cases. The RCS cold-leg temperature (T_{cold}), defined by the vessel inlet (RCP outlet) temperature on the PCWG parameters for the Kewaunee 7.4-percent Power Uprating Program, is a maximum of 539.2°F. The maximum uprating RCS T_{cold} is less than the corresponding RSG Project T_{cold} temperature of 543.8°F, or the lowest temperature considered in the original analyses, 552°F. Since none of the temperatures exceeds the previously considered temperatures, and the pressure does not change, the uprating PCWG parameters are bounded by those used as inputs to the original and RSG analyses.

Table 5.6-1 summarizes the cold-leg PCWG temperatures. From Table 5.6-1, the present (RSG) conditions provide an RCS T_{cold} range of 521.9° to 543.8°F, compared to an RCS T_{cold} range of 521.9° to 539.2°F for the Power Uprating Program. Therefore, the present operation range bounds the RCS T_{cold} range for the Power Uprating Program.

5.6.1.3.2 Transient Discussion

Several of the cold-leg transients defined for the Power Uprating Program have small increases in the temperature change associated with the transient, when compared to the NSSS cold leg design transients defined for the Kewaunee RSG Project. These transients are summarized in Table 5.6-2. The unit-loading/unloading, loss-of-load, loss-of-power, loss-of-flow, and reactor-trip-from-full-power transients all show increases in the amount of temperature change. In no case does the increased temperature change differ from the temperature change defined for the RSG Project by more than 1.3°F. The largest temperature change associated with any of these transients is 47.1°F for the loss-of-power transient. These small changes are acceptable for the Power Uprating Program.

There are no changes in the pressure transients associated with these system transients.

5.6.1.4 Acceptance Criteria

The acceptance criteria for the ASME Code structural analysis of the RCP pressure boundary are that the analyzed stresses do not exceed the stress allowables of the ASME Code and that the cumulative usage factors from the Code fatigue analysis remain less than 1.0.

When the RSG evaluation was performed, no increases were required in the stresses that had been determined in the original generic stress analyses. This is because the stresses associated with the temperature and pressure changes considered in the original analyses bounded the stresses associated with temperature and pressure changes defined for the RSG Project. This occurs because, in the original analyses, the design transients were grouped and only the most severe transient in each group was analyzed. The total number of transient cycles for all transients in the group was considered for each group, along with the worst-case stresses. This was done to reduce the number of analyses required.

The cumulative usage factors were increased where changes to the design transients indicated that an increase was necessary. The fatigue usage factors were increased for the RSG Project due to the addition of two transients that were not considered in the original analyses. However, where changes to the design transients would have allowed a decrease in stresses or cumulative usage factors, no decrease was calculated, and no credit was taken for such a decrease. Therefore, the RCP components are still acceptable for the new NSSS design transients if the new NSSS design transients are bounded by either the original NSSS design transients, or the NSSS design transients used in the RSG evaluation. If the NSSS design transients are shown to be bounded by those considered for either the RSG Project or the original design, then:

- The cumulative usage factors calculated for the RCPs for the RSG Project remain bounding and applicable for the Power Uprating Program.
- The stresses calculated in the original design analyses remain bounding and applicable for the Power Uprating Program.

Since the RCPs were acceptable for the RSG Project, the RCPs remain acceptable for the Kewaunee Power Uprating Program.

5.6.1.5 Results

The operating temperature and pressure discussion presented above in subsection 5.6.1.3.1 showed that the operating pressures and temperatures are bounded by those considered for the RSG Project.

The differences between the previous (RSG) evaluation and the current Power Uprating Program transients, as discussed in subsection 5.6.1.3.2, are slight modifications to the temperature changes associated with a few of the NSSS design transients which are acceptable.

The temperature transients that show slight increases in the temperature differences for the Power Uprating Program, shown in Table 5.6-2, are all transients that were considered in the same grouping in the original design reports. The applicable transients used in the original design reports to represent this transient group considered a temperature change of at least 52.5°F. This is larger than the maximum temperature change of 47.1°F defined for any of the Power Uprating Program transients that showed an increase in temperature change from the RSG Project transients. Since the current Power Uprating Program PCWG parameters and the NSSS design transients are all bounded by the original evaluation inputs or by the RSG evaluation inputs, the Power Uprating Program operating parameters are acceptable for the RCPs.

As a summary of the analysis performed for the RSG Project, the cumulative usage factors that were recalculated or verified for the RSG Project are presented in Table 5.6-3. It is noted that the fatigue waiver, allowed by Article 4, paragraph N-415.1, of the 1968 Code (Reference 2) or by NB-3222.4 (d) of the 1971 Code, is cited for many of the components of the RCPs. This indicates that, by the Code rules, a detailed fatigue calculation leading to a value of the cumulative usage factor is not required for those components.

5.6.1.6 Conclusions

The Power Uprating Program PCWG parameters and NSSS design transients have been shown to be bounded by the parameters and transients considered for either the RSG Project or the original design analyses. Therefore, the conclusions of the RSG Project evaluation are still valid and applicable to the Power Uprating Program. The RCPs are acceptable from a structural standpoint. The RCP pressure boundary parts still comply with the ASME Code originally specified or later editions. Therefore, the evaluation results of the Power Uprating Program for the RCPs are consistent with and continue to comply with the current licensing basis/acceptance requirements for Kewaunee.

5.6.2 Reactor Coolant Pump Motors

5.6.2.1 Introduction and Background

This section addresses the performance of the reactor coolant pump (RCP) motors. The RCP motors are evaluated for the Kewaunee Power Uprating Program Performance Capability Working Group (PCWG) parameters (Section 2) and best-estimate flows, which assumed a 7.4-percent increase in core power.

5.6.2.2 Input Parameters and Assumptions

The input parameters considered in the evaluation of the RCP motors are the steam generator outlet temperatures (Section 2) and the flows defined for the Kewaunee Nuclear Power Plant (KNPP) 7.4-percent Power Uprating Program. These parameters are considered for the Kewaunee Model 93A RCPs consisting of the two original pumps supplied to Kewaunee and the spare pump supplied to Kewaunee.

5.6.2.3 Description of Analysis

The steam generator outlet temperatures and best-estimate flows are considered in a hydraulic analysis using the operating characteristics of the Kewaunee reactor coolant pumps. This hydraulic analysis calculates the power requirements for the impeller that operates at the highest cold power. For the Kewaunee 7.4-percent Power Uprating Program, the power requirements from this analysis for hot-loop and cold-loop operation were compared to the power requirements considered for the Kewaunee Replacement Steam Generator (RSG) Project evaluation. Based on a negligible increase in the power requirements between the Power Uprating Program and the RSG Project, the RCP motors are considered acceptable for the Power Uprating based on the RSG Project evaluation.

The RSG Project evaluated the RCP motor loading in four areas:

- Continuous operation at hot-loop temperatures and flows
- Continuous operation at cold-loop temperatures and flows
- Starting across the line with a minimum 80-percent starting voltage
- Loads on the thrust bearings

5.6.2.4 Acceptance Criteria

For the Kewaunee Power Uprating Program, the acceptance of the RCP motor loading is based on the change from the loading previously evaluated in the RSG Project being negligible. The acceptance criteria used for evaluating the motor loading for the RSG Project were taken from the Equipment Specifications for the motor (References 7 and 8).

Per the Equipment Specifications, the motor is required to drive the pump continuously under hot-loop conditions without exceeding a stator winding temperature rise of 75°C. This corresponds to the National Electric Manufacturers' Association (NEMA) Class B temperature rise limit in a 50°C ambient temperature.

Per the Equipment Specification (Reference 7), the motor is required to drive the pump for up to 50 hours (continuous) and 3000 hours maximum over the 40-year design life under cold-loop conditions without exceeding a stator winding temperature rise of 100°C. This corresponds to the NEMA guaranteed limit for a Class F winding in a 50°C ambient temperature.

Per the Equipment Specification (Reference 8), the motor is required to start across the line with a minimum 80-percent starting voltage against the reverse flow of the other pump running at full speed under cold-loop conditions. The limiting component for this type of loading is the rotor cage winding, which has design limits of a 300°C temperature rise on the bars and 50°C temperature rise on the rings.

The thrust-bearing loading used for the motor design is given in the Equipment Specification (Reference 8). Performance of the thrust bearings in an RCP motor can be adversely affected by excessive or inadequate loading. The thrust-bearing loading for the revised conditions is compared to the design thrust-bearing loading to determine continued acceptability.

5.6.2.5 Results

The worst case loads for the RCP motors were calculated for the Kewaunee Power Uprate Program operating conditions. The new worst-case hot-loop load under the revised operating conditions is 5942 HP. The new worst-case cold-loop load under the revised operating conditions is 7656 HP. These loadings are not significantly different from the motor loadings of 5940 HP for hot-loop operation, and 7653 HP for cold-loop operation that were previously evaluated for the RSG Project. Thus, the revised motor loadings are acceptable based on the previous evaluation.

The evaluations of the RCP motors that were the basis of the RSG Project conclusions are described in the following paragraphs.

Continuous Operation at Hot-Loop Conditions

The worst-case hot-loop operating load for the RSG Project was 5940 HP, which is below the nameplate rating of the motor, 6000 HP. Since the loading is within the nameplate rating of the motor, it is acceptable without further calculations. The Power Uprating Program hot-loop loading of 5942 HP is also less than the nameplate rating of 6000 HP.

Continuous Operation at Cold-Loop Conditions

The worst-case cold-loop operating load of 7653 for the RSG Project exceeded the nameplate cold-loop rating of the motor, 7500 HP, by 2 percent. Testing on duplicate motors has shown a stator temperature rise no greater than 77.7°C at the cold-loop nameplate rating of 7500 HP.

Analysis indicated that the cold-loop temperature rise of the stator at the RSG Project loading of 7653 will be approximately 79°C, which is well below the NEMA limit given in subsection 5.6.2.4. The cold-loop loading for the Power Uprating Program is 7656 HP, an increase of less than 0.04 percent from the RSG Project loading. The stator temperature will thus also remain well below the NEMA limit for the Power Uprating Program conditions.

Starting

The starting temperature rise of the rotor cage winding for the RSG conditions was calculated using a conservative all-heat-stored analysis. The results of that analysis indicated temperature rises of 226.2° and 37.21°C, respectively, for the rotor bars and the resistance rings. These did not exceed the design limits given in subsection 5.6.2.4. The temperature rise with the insignificant changes in motor loading associated with the Power Uprating Program will likewise not exceed the design limits.

Thrust-Bearing Loading

The thrust-bearing loadings for the RSG Project conditions indicated a reduction in the thrust-bearing load of 1923 lbs for hot-loop operation, and a reduction of 4578 lbs for cold-loop operation. In comparison to the normal operating thrust-bearing load of 104,400 lbs given in the Equipment Specification (Reference 8), these changes were not considered significant and the thrust bearings were considered acceptable for the RSG Project loads. The insignificant changes in motor loading associated with the Power Uprating Program are also considered acceptable for the thrust bearings.

5.6.2.6 Conclusions

The RCP motors were previously evaluated in four areas for the RSG Project conditions under loadings of 5940 HP for worst-case hot-loop operation, and 7653 HP for worst-case cold-loop operation. Since the new RCP motor loads show no significant change from the bounding loads considered in the RSG analysis, no further evaluation is required. The RCP motors at the KNPP are still considered acceptable for the Power Uprate Program conditions.

5.6.3 References

1. ES 676557, *Kewaunee Controlled Leakage Pump Assembly*, Rev. 3, Westinghouse Electric Corporation, Nuclear Energy Systems, Pittsburgh, PA, February 7, 1975.

2. *ASME Boiler and Pressure Vessel Code, Section III*, "Rules for Construction of Nuclear Vessels," 1968 Edition, and later Editions and Addenda, The American Society of Mechanical Engineers, New York.

3. Engineering Memorandum 4487, *Stress Analysis of the Casing, Main Flange, Main Flange Bolts, and Thermal Barrier of the 93A Shaft Seal Pump*, Rev. 1, J. D. Nee, Westinghouse Electro-Mechanical Division, August 16, 1974.

4. Engineering Memorandum 4528, *Analysis of the 93A Casing Nozzles Using Umbrella Loads*, Rev. 3, R. J. Oleyar, Westinghouse Electro-Mechanical Division, April 21, 1976.

5. Engineering Memorandum 4503, *Analysis of the 93A Casing Feet Using Umbrella Loads*, Rev. 1, R. J. Oleyar, Westinghouse Electro-Mechanical Division, February 8, 1974.

6. Engineering Memorandum 4895, *Stress Analysis of 93A Pump Non-Pressure Components for Wisconsin Public Service Corporation Kewaunee Plant Shop Order U311*, P. J. Cronin, Westinghouse Electro-Mechanical Division, October 4, 1976.

7. ES E-565614, Revision Q, *General Specification for Induction Motor for Shaft Seal Type Pump*, Westinghouse Electric Corporation, Atomic Equipment Division, August 9, 1972.

8. ES E-565626, *Supplementary Ordering Information for Shaft Seal-Type Pump Motor*, Rev. D, Westinghouse Electric Corporation, Atomic Equipment Division, January 8, 1971.

Table 5.6-1 **PCWG Conditions Used to Bracket All Operating Conditions for Kewaunee 7.4-Percent Power Uprating Program**				
Parameter	**Present**		**7.4-Percent Uprating Program**	
	High T_{avg}	**Low T_{avg}**	**High T_{avg}**	**Low T_{avg}**
T_{cold} (vessel inlet)	543.8°F	521.9°F	539.2°F	521.9°F

Table 5.6-2 **Cold Leg Thermal Transient Summary for RCP Evaluation for Kewaunee 7.4-Percent Power Uprating Program**				
	7.4-Percent Uprating Program		**RSG Project[1]**	
	Max ΔT (°F) High T_{avg}	**Max ΔT (°F) Low T_{avg}**	**Thermal Transient (°F)**	**Max ΔT (°F)**
Normal Condition				
Unit Loading/Unloading	8.1	25.4	546.5-522-546.5	24.5
Upset Condition				
Loss of Load	37.3	44.4	521.9-565-560 543.8-581.1-560.5 (See note 1)	43.1 37.3 (See note 1)
Loss of Power	22.2	47.1	521.9-568-558	46.1
Loss of Flow (no RCP)	18.8	30.4	522-517-547	30.0
Loss of Flow (w/RCP)	14.6	25.7		
Reactor Trip from Full Power	13	5	543.8-555-543.3	11.7

Notes:

1. These transients include both the high- and low-operating temperatures within the RCS T_{avg} window, with the exception of the loss-of-load transient. For the loss-of-load transient, the first transient listed is for the high-temperature case, and the second transient listed is for the low-temperature case.

Table 5.6-3

RCP Fatigue Evaluation for Kewaunee RSG Project and Kewaunee 7.4-Percent Power Uprating Program

RCP Component	Cumulative Usage Factor from Original Analyses	Cumulative Usage Factor RSG Project and 7.4-% Power Uprating Program
Casing	Fatigue Waiver	Fatigue Waiver
Main Flange	Fatigue Waiver	Fatigue Waiver
Thermowell	Negligible	[ ]^{a,c}
Main Flange Bolts	[ ]^{a,c}	[ ]^{a,c}
Thermal Barrier Flange	[ ]^{a,c}	[ ]^{a,c}
Suction/Discharge Nozzles	Fatigue Waiver	Fatigue Waiver
Casing Feet	Fatigue Waiver	Fatigue Waiver

Bracketed [ ]^{a,c} information designates data that is Westinghouse Proprietary, as dicussed in Section 1.7 of this report.

5.7 Steam Generator Component Evaluations

Kewaunee Nuclear Power Plant (KNPP) has proposed uprating their operating Nuclear Steam Supply System (NSSS) power to 1780 MWt. This represents a power uprating of 7.4 percent. To support the planned 7.4-percent power uprating, the existing Model 54F replacement steam generators (RSGs) have been evaluated for operation at the uprated power condition. 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 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 condition 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 conditions. Detailed discussions regarding the evaluations and the conclusions reached for each aspect of steam generator operation are included in specific subsections within Section 5.7 of this report.

5.7.1.1 Thermal-Hydraulic Evaluation

This section summarizes the thermal-hydraulic analyses performed to determine the operating characteristics of the KNPP. 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 assessments are made:

- Steam generators will be hydro-dynamically stable; the damping factor is highly negative.
- MCO remains below the design limit of 0.25 percent for all cases analyzed.

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

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

5.7.1.1.1 Input Parameters and Assumptions

Operating Conditions

The reference case (referred to as Case 0) for this evaluation was established as the 100-percent power case (nominal 1650-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 7.4-percent uprated power design parameters are defined in Section 2 of this document. Cases 1 through 4 are evaluated for the 107.4-percent NSSS power of 1780 MWt, and correspond to the four operating cases shown in Section 2. 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.

Assumptions

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.

5.7.1.1.2 Description of Analysis and 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. Case 0 states again that 100-percent power is nominal 1650-MWt NSSS power.

The GENF calculated operating conditions were utilized to calculate the 3-D flow field parameters on the secondary side of the steam generator. A separate 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.

Case 1 – 7.4-percent uprate plant parameters with the steam generator inlet temperature of T_{hot} = 590.8°F and 0-percent tube plugging from Section 2.

Case 2 – 7.4-percent uprate plant parameters with steam generator inlet temperature of T_{hot} = 590.8°F and 10-percent tube plugging from Section 2.

Case 3 – 7.4-percent uprate plant parameters with the steam generator inlet temperature of T_{hot} = 606.8°F and 0-percent tube plugging from Section 2.

Case 4 – 7.4-percent uprate plant parameters with the steam generator inlet temperature of T_{hot} = 606.8°F and 10-percent tube plugging from Section 2.

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 moisture carryover (MCO) may result in erosion-corrosion problems in the steam piping and/or steam turbine. Prior MCO assessments 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 RSG MCO assessment also includes field data and correlations between the 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(specific volume 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 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 results. 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 model is determined. 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 a 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 percent or less.
- The variations in the secondary fluid flow rates and velocities are also small: approximately 12 percent 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 Model 54F RSGs focused on the changes to secondary side operating characteristics at the 7.4-percent power 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 analyses show that the maximum (X/X_{DNB}) increases from [ ]^{a,c} at 100-percent power, to [ ]^{a,c} 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. The KNPP Model 54F RSGs include the same Moisture Separation System improvements as the Model 51 units tested.

The test results indicated that the separation system improvements are highly effective. The calculated carryover was [ ]^{a,c} 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 WCAP-15559. The calculated MCO increased from [ ]^{a,c} percent of steam flow at 100-percent power, to [ ]^{a,c} percent at 107.4-percent power uprate conditions, with 0-percent tube plugging. With 10-percent tube plugging, the maximum calculated MCO was [ ]^{a,c} percent. All calculated MCO values are below the 0.25-percent limit at the 7.4-percent power uprate condition.

Hydro-Dynamic Stability

The hydro-dynamic 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 numerically (actually an increase in damping) from [ ]^{a,c} at nominal power, to a minimum value of [ ]^{a,c}, 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 [ ]^{a,c} (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 [ ]^{a,c} psi to [ ]^{a,c} psi as a result of a 7.4-percent power uprate. It further increased to [ ]^{a,c} 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 [ ]^{a,c} psi with 0-percent tube plugging, and [ ]^{a,c} 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 [ ]^{a,c} to [ ]^{a,c} 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 RSGs are projected to 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 replacement steam generators (RSGs) at the KNPP. The Kewaunee power uprate evaluation was based on the existing analyses and evaluations from the previous RSG Project.

The Performance Capability Working Group (PCWG) operating parameters for the Kewaunee 7.4-percent power uprate conditions are summarized in Section 2 of this report. For the RSG analyses in support of the uprating, the previous RSG design transients should be used for the 7.4-percent power uprate, except for the feedwater-related design parameters (temperature and flow rate). On this basis, 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 power uprate required further evaluation. The affected analyses include the feedwater nozzle and thermal sleeve analysis and the J-nozzle-to-feedring weld fatigue analysis. The feedring seismic and steam line break analysis is not affected by the T_{feed} changes associated with the uprate. 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. 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 steam generator replacement.

5.7.2.1 Input Parameters and Assumptions

The PCWG parameters applicable to the Kewaunee 7.4-percent Uprate Project have been defined in Section 2. A review of the applicability of the existing RSG analyses has been performed in consideration of the NSSS design parameters specified in Section 2. 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 previously 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 previous 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 specified 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, and the feedring-to-J-nozzle weld fatigue analysis.

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. The structural integrity of the feedwater nozzle and thermal sleeve and the fatigue analysis of the J-nozzle-to-feedring weld have been evaluated 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 is 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° 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° 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, to demonstrate the acceptability of the maximum ranges of stress intensity ranges at critical analysis sections 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 previously developed 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 satisfy the ASME Code (Reference 2) specified allowables. In addition, the cummulative fatigue usage factor must be less than or equal to unity.

For certain sections in the original feedwater nozzle and thermal sleeve analysis the maximum range of stress intensity already exceeded the Code allowable of $3S_m$. For those cases, the appropriate K_e factors were developed 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 locations, 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 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 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 value 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 2).

Of the limiting stress locations analyzed in the original RSG feedwater nozzle and thermal sleeve analysis, most of these locations had low cumulative fatigue usage factors (CUFs) ranging from [ ]^{a,c}. The most limiting fatigue usage factors were [ ]^{a,c}. 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 [ ]^{a,c}. Therefore, it is concluded that for all of the locations evaluated, the CUFs for the uprated conditions satisfy the CUF requirements of the ASME Code.

For the J-nozzle-to-feedring weld, review of the structural analysis 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. The maximum stress intensity ranges at the limiting locations 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 decreased slightly, leading to a small increase in the stress intensity range for load combinations involving this event.

The fatigue calculations 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 increased only slightly, from a value of [ ]^{a,c} to a value of [ ]^{a,c} 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 CUFs 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 power 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 1800 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 Section 2, 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 (Reference 1).

Similarly, the transient parameters applicable to the Power Uprate Project are defined in Section 3 of this report.

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 and 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 1800 psi, as defined in the applicable design specification (Reference 1). The design pressure requirements for Class 1 equipment are defined in the applicable edition of the ASME Boiler and Pressure Code (B&PV) Section III (Reference 2) for the Kewaunee steam generators, 1986 through 1987 addenda. 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 1800 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 1980 psi.

5.7.3.4 Results

The maximum primary-to-secondary differential pressures for high T_{ave} conditions were found to be 1552 psi and 1511 psi for normal and upset conditions, respectively. For low T_{ave} conditions, the maximum pressure differentials are 1630 psi, and 1605 psi for normal and upset conditions, respectively. These values were all below the applicable design pressure limits of 1800 psi for normal conditions, and 1980 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 power uprate.

5.7.4 Tube Vibration and Wear

The impact of the proposed 7.4-percent power 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 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 [ ]^{a,c}, 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 Power 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 predicts 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 would not result in unacceptable wear.

The analysis of the effects of the Uprated Power condition on the steam generator tubes has addressed both the straight leg and U-bend portions of the steam generator. 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 percent over the current levels of wear. Since significant tube wear is not currently occurring at Kewaunee, a 25-percent increase will not be significant. Should significant tube wear begin sometime in the future, the rate of tube wear would be sufficiently small so 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 steam generator tubes respond primarily to the conditions associated with the secondary side since the forcing functions associated with the secondary side of the steam generator 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 steam generator 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 uppermost plate. However, since the Kewaunee steam generator tube support plates are manufactured from stainless steel, there is no potential for the necessary boundary conditions (that is, 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-inch 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-inch wall thickness tube undercut (40-percent wall reduction) is considered and the reduced weld joint geometry qualified.

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 [ ]^{a,c} psi. This parameter remains unchanged following the power uprate.

5.7.6.1 Input Parameters and Assumptions

The mechanical plug evaluated for Kewaunee is the Westinghouse ribbed mechanical plug, 7/8-inch diameter, long and short lengths. The plug material is SB-166, Alloy 690, as described in ASME Code Case N-474-1 (Reference 3).

The tube material for the Kewaunee RSG tubes is SB-163, Alloy 690, as defined in ASME Code Case N-20-3 (Reference 4).

5.7.6.2 Description of Analysis and Evaluation

A structural evaluation was performed for the Westinghouse mechanical plug for the 7.4-percent power uprate condition. This evaluation was performed in accordance with the applicable requirements of the ASME B&PV Code (Reference 2). 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 2485 psi ΔP design condition. This design ΔP bounds all maximum normal and upset differential pressures calculated for the 7.4-percent Power Uprating. The primary-to-secondary differential is actually limited to 1800 psi for normal design conditions. The 1800 psi value is well below the design pressure of 2485 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 would be fabricated to the requirements of a more recent ASME Code Edition. On this basis, the evaluation also determined that the mechanical plug is acceptable for the 7.4-percent power uprate, based on the 1989 ASME Code Edition requirements (Reference 5).

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 2).

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 2485 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. 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 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 weld process 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 (Reference 3). 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 (Reference 4).

5.7.7.2 Description of Analysis and Evaluation

A structural evaluation was performed for the existing field 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 2).

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 1800 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 1700 psi. This was the controlling pressure condition for the baseline transient conditions. The governing differential pressure for the 7.4-percent power uprate for normal/upset conditions was calculated at

1630 psi, which is bounded by the baseline 1700 psi differential pressure. It was shown that the stress limits are acceptable for a 1700-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 field 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 2). 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 [ ]^{a,c} 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 design 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 inch 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 (Reference 4).

5.7.8.2 Description Of Analysis and Evaluation

A structural evaluation was performed for the undercut of the steam generator tube ends for the 7.4-percent power uprate condition. The evaluation was performed to the applicable requirements of ASME B&PV Code (Reference 2). 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 power uprate 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. 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 power uprate condition are all within ASME Code allowable values. The maximum range for stress intensity was

[ ]a,c, 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 power uprate, remain acceptable. The maximum cumulative fatigue usage factor was calculated as [ ]a,c, which remains less than the allowable factor of unity.

5.7.8.5 Conclusions

The 7.4-percent power uprate stress evaluation of tube undercut in the Kewaunee Model 54F steam generators determined that the stresses are all within ASME Code allowable values (Reference 2). The fatigue usage values were found to be less than 1.0, therefore a 0.020-inch tube wall thickness undercut is acceptable for operation at the 7.4-percent power uprate 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 (Reference 6).

5.7.10 Tube Repair Limits (Regulatory Guide 1.121 Analysis)

The heat transfer area of steam generators in a pressurized-water reactor (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 7) 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 has been 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. These original generators were replaced in late 2001 with Westinghouse Model 54F steam generators. The new steam generators use Alloy 690 thermally treated tubes.

5.7.11.1 Input Parameters and Assumptions

As stated above, the Kewaunee plant uses Westinghouse Model 54F RSGs. The Model 54F steam generator has 3592 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. Stainless steel TSP construction precludes classical "denting" observed in plants with carbon steel TSPs. The row 1 U-bend minimum bend radius is 3.141 inch, which is greater than the original steam generator row 1 U-bend minimum bend radius of 2.19 inch. The larger bend radius reduces residual stresses from fabrication. Additionally, the first nine rows of tubes in the Kewaunee Model 54F RSGs 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 (that is, negligible). 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 enhanced corrosion resistance of the Alloy 690 thermally treated material, coupled with the supplemental thermal treatment following bending, is expected to result in a condition such that the corrosion susceptibility of the small radius U-bends is not elevated in comparison to the straight length regions.

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 and 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. A Westinghouse report (Reference 9) 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.

The Westinghouse report (Reference 9) 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. This report also 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 secondary side water chemistry control consistent with the EPRI Secondary Water Chemistry Guidelines (Reference 10). ODSCC at the top of tubesheet region is addressed partly through the use of Alloy 690 thermally treated tube material, hydraulic tube expansion, and 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, with hydraulic expansion residual stress more uniformly distributed in comparison to explosive expansion processes. The uniformity of the stress resulting from hydraulic expansion will result in a less ODSCC-susceptible condition.

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.

Future steam generator tube degradation will be addressed through the condition-monitoring and operational-assessment process, as described in the EPRI Tube Integrity Guideline (Reference 8). The only tube-degradation mechanisms that may be affected in the near term by the 7.4-percent Power 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 Power 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 nondestructive examination (NDE). The growth rate of the reported TSP wear indications was zero.

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.

5.7.12 References

1. Design Specification 414A03, *Wisconsin Public Service Corporation Kewaunee Nuclear Power Plant, Model 54F Replacement Steam Generator Complete Lower Assembly and Modified Upper Assembly*, Rev. 6, April 17, 2001.

2. ASME Boiler and Pressure Vessel Code, *Rules for Construction of Nuclear Power Plant Components*, Section III, 1986 Edition, 1987 Addenda (applies to RSGs), The American Society of Mechanical Engineers, New York, NY.

3. ASME Code Case N-474-1, *Design Stress Intensities and Yield Strength Values for UNS N06690 with a Minimum Specified Yield Strength of 35 ksi, Class 1 Components, Section III, Division 1.* Approval Date: March 5, 1990.

4. ASME Code Case N-20-3, *SB-163 Nickel-Chromium-Iron Tubing (Alloys 600 and 690) and Nickel-Iron-Chromium Alloy 800 at a Specified Minimum Yield Strength of 40.0 ksi and Cold Worked Alloy 800 at a Yield Strength of 47.0 ksi*, Section III, Division 1, Class 1. Approval Date: November 30, 1988.

5. ASME B&PV Code Section III, *Rules for Construction of Nuclear Power Plant Components*, American Society of Mechanical Engineers, New York, NY, 1989.

6. WCAP-15941, *Kewaunee Model 54F Steam Generator Parametric Loose Object Evaluation Licensing Report*, November 2002.

7. NRC Regulatory Guide 1.121, *Bases for Plugging Degraded PWR Steam Generator Tubes (for comment)*, August 1976.

8. EPRI TR-107621-R1, *Steam Generator Integrity Assessment Guidelines: Revision 1*, March 2000.

9. LTR-CDME-02-35, *An Assessment of the Projected Performance of Model D5, F, and Advanced Steam Generators with Thermally Treated alloy 600 and Alloy 690 Heat Transfer Tubing*, February 2002 (Proprietary).

10. EPRI TR-102134-R5, *PWR Secondary Water Chemistry Guidelines: Rev. 5*, March 2000.

Table 5.7-1

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

Operating Conditions	Case 0	Case 1	Case 2	Case 3	Case 4
Power, %	100	107.4	107.4	107.4	107.4
Reactor Power, MWt	1650	1772	1772	1772	1772
NSSS Power, MWt	1658	1780	1780	1780	1780
Power per Steam Generator, MWt	829	890	890	890	890
Primary Flow per Loop, GPM	89,000	89,000	89,000	89,000	89,000
Primary Fluid Pressure, psia	2250	2250	2250	2250	2250
Primary Hot-Leg Temperature, °F	586.3	590.8	590.8	606.8	606.8
Feedwater Temperature, °F	427.5	437.1	437.1	437.1	437.1
Water Level Above Tubesheet, inch	493.6	493.6	493.6	493.6	493.6
Blowdown Flow Rate, lb/hr	36,000	36,000	36,000	36,000	36,000
Tube Plugging, % of Total Tubes	0	0	10	0	10
Thermal-Hydraulic Characteristics					
Steam Flow Rate per Steam Generator, 10^6 lb/hr					
Steam Pressure at Nozzle, psia					
T_{steam} at Nozzle, °F					
Circulation Ratio					
Separator Parameter, 10^{12} lb-ft^3/hr^2					
MCO, %					
Downcomer Fluid Velocity, ft/sec					
Sec. Side Pressure Drop, psi					
Sec. Fluid Liquid Mass, lb					
Sec. Fluid Liquid Volume, ft^3					
Sec. Fluid Vapor Mass, lb					
Sec. Fluid Vapor Volume, ft^3					
Sec. Fluid Heat Content, 10^6 BTU					
Average Heat Flux, BTU/hr-ft^2					
Damping Factor,1/hr					
Steam Generator Average, T_{avg} – Temperature, °F					
Steam Generator Outlet, T_{cold} – Temperature, °F					
Maximum (X/X_{DNB})					

a,c

Bracketed [ ]^{a,c} information designates data that is Westinghouse Proprietary, as dicussed in Section 1.7 of this report.

Table 5.7-2

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

Analysis Section	Surface	Event Combination (from Table 8-8 of Ref. 17)	Load Combination Description	Max. Range of SI (ksi, from Table 8-8 of Ref. 17)	Uprated Condition Max Range of SI (ksi) (1)	ASME Code Limit
ASN 2	Inside	1-1 to 3-4	Cold Shutdown – Plant Load/Unload	[ ][a,c]	[ ][a,c]	69.90
	Outside	1-1 to 10-1	Cold Shutdown – OBE (Max Stress)	[ ][a,c]	[ ][a,c]	61.38
ASN 5	Inside	1-1 to 3-4	Cold Shutdown – Plant Load/Unload	[ ][a,c]	[ ][a,c]	69.90
	Outside	1-1 to 10-1	Cold Shutdown – OBE (Max Stress)	[ ][a,c]	[ ][a,c]	90.00
ASN 7	Inside	1-1 to 16-1	Cold Shutdown – Therm. Strat. Prof. 1	[ ][a,c]	[ ][a,c]	90.00
	Outside	7-10 to 14-1	Hot Standby – Class 1 Leak Test	[ ][a,c]	[ ][a,c]	90.00
ASN 8	Inside	1-1 to 7-27	Cold Shutdown – Hot Standby	[ ][a,c]	[ ][a,c]	80.10
	Outside	1-1 to 7-27	Cold Shutdown – Hot Standby	[ ][a,c]	[ ][c]	80.10
ASN 9	Inside	to 12-2 1-1 to 7-27	Cold Shutdown – Secondary Hydro. Cold Shutdown – Hot Standby	[ ][a,c]	[ ][a,c]	80.10 80.10
	Outside	1-1 to 9-13	Cold Shutdown – Loss of Load	[ ][a,c]	[ ][a,c]	80.10
ASN 10	Inside	1-1 to 12-2	Cold Shutdown – Secondary Hydrotest	[ ][a,c]	[ ][a,c]	80.10
	Outside	1-1 to 37-1	Cold Shutdown – Therm. Strat. Prof. 6	[ ][a,c]	[ ][a,c]	80.10
ASN 12	Inside	1-1 to 7-10	Cold Shutdown – Hot Standby	[ ][a,c]	[ ][a,c]	69.90
	Outside	3-2 to 7-11	Plant Load/Unload – Hot Standby	[ ][a,c]	[ ][a,c]	90.00
ASN 14	Inside	9-11 to 16-1	Loss of Load – Therm. Strat. Prof. 1	[ ][a,c]	[ ][a,c]	69.90
	Outside	3-3 to 14-1	Plant Load/Unload – Class 1 Leak Test	[ ][a,c]	[ ][a,c]	69.90
ASN 15	Inside	9-11 to 16-1	Loss of Load – Therm. Strat. Prof. 1	[ ][a,c]	[ ][a,c]	69.90
	Outside	1-1 to 9-11	Cold Shutdown – Loss of Load	[ ][a,c]	[ ][a,c]	69.90

Table 5.7-2 (continued)

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

Notes:

1. 1.04 x (max range of SI from Table 8-8 of Reference 17), unless otherwise indicated.
2. The stress meets the S_y limit of 95.52 ksi (NB-3226 of the ASME Code) for stress range combinations that include at least one extreme caused by test conditions. All other stress ranges are less than the $3S_m$ limit of 80.10 ksi. As a test condition, this stress range is unaffected by the increased feedwater temperature at uprated conditions.
3. As a test condition, this stress range is unaffected by the increased feedwater temperature at uprated conditions. All other event combinations at this ASN produce stress ranges that are well below $3S_m$ for the nominal and uprated conditions.
3. This range is acceptable based on the simplified elastic-plastic analysis procedure given in NB-3228.5 of the ASME Code. An appropriate penalty factor is used in the fatigue evaluation for the inside surface of ASNs 13-15 (a fatigue evaluation is not performed for ASN 12). All other stress ranges at this ASN are less than the $3S_m$ limit of 69.90 ksi.

Bracketed [ ]^{a,c} information designates data that is Westinghouse Proprietary, as dicussed in Section 1.7 of this report.

Table 5.7-3

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

Analysis Section	Surface	Event Combination (from Table 8-8 of Ref. 17)	Load Combination Description	Max. Range of SI (ksi, from Table 8-8 of Ref. 17)	Uprated Condition Max Range of SI (ksi) (1)	ASME Code Limit
ASN 2	Inside	1-1 to 3-4	Cold Shutdown – Plant Load/Unload	[ ]^{a,c}	[ ]^{a,c}	69.90
	Outside	1-1 to 10-1	Cold Shutdown – OBE (Max Stress)	[ ]^{a,c}	[ ]^{a,c}	61.38
ASN 5	Inside	1-1 to 3-4	Cold Shutdown – Plant Load/Unload	[ ]^{a,c}	[ ]^{a,c}	69.90
	Outside	1-1 to 10-1	Cold Shutdown – OBE (Max Stress)	[ ]^{a,c}	[ ]^{a,c}	90.00
ASN 7	Inside	1-1 to 10-1	Cold Shutdown – OBE (Max Stress)	[ ]^{a,c}	[ ]^{a,c}	90.00
	Outside	7-10 to 14-1	Hot Standby – Class 1 Leak Test	[ ]^{a,c}	[ ]^{a,c}	90.00
ASN 8	Inside	1-1 to 7-27	Cold Shutdown – Hot Standby	[ ]^{a,c}	[ ]^{a,c}	80.10
	Outside	1-1 to 22-1	Cold Shutdown – Therm. Strat. Prof. 3	[ ]^{a,c}	[ ]^{a,c}	80.10
ASN 9	Inside	1-1 to 12-2	Cold Shutdown – Secondary Hydro. Cold Shutdown – Hot Standby	[ ]^{a,c}	[ ]^{a,c}	80.10 80.10
	Outside	1-1 to 7-27	Cold Shutdown – Hot Standby	[ ]^{a,c}	[ ]^{a,c}	80.10
ASN 10	Inside	1-1 to 12-2	Cold Shutdown – Secondary Hydrotest	[ ]^{a,c}	[ ]^{a,c}	80.10
	Outside	1-1 to 37-1	Cold Shutdown – Therm. Strat. Prof. 6	[ ]^{a,c}	[ ]^{a,c}	80.10
ASN 12	Inside	1-1 to 7-10	Cold Shutdown – Hot Standby	[ ]^{a,c}	[ ]^{a,c}	69.90
	Outside	3-2 to 7-11	Plant Load/Unload – Hot Standby	[ ]^{a,c}	[ ]^{a,c}	90.00
ASN 14	Inside	9-11 to 14-1	Loss of Load – Class 1 Leak Test	[ ]^{a,c}	[ ]^{a,c}	69.90
	Outside	3-3 to 14-1	Plant Load/Unload – Class 1 Leak Test	[ ]^{a,c}	[ ]^{a,c}	69.90
ASN 15	Inside	1-1 to 9-11	Cold Shutdown – Loss of Load	[ ]^{a,c}	[ ]^{a,c}	69.90
	Outside	1-1 to 9-11	Cold Shutdown – Loss of Load	[ ]^{a,c}	[ ]^{a,c}	69.90

Table 5.7-3 (continued)

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

Notes:

1. 1.04 x (max range of SI from Table 8-8 of Reference 17), unless otherwise indicated.
2. The stress meets the S_y limit of 95.52 ksi (NB-3226 of the ASME Code) for stress range combinations that include at least one extreme caused by test conditions. All other stress ranges are less than the $3S_m$ limit of 80.10 ksi. As a test condition, this stress range is unaffected by the increased feedwater temperature at uprated conditions.
3. As a test condition, this stress range is unaffected by the increased feedwater temperature at uprated conditions. All other event combinations at this ASN produce stress ranges that are well below $3S_m$ for the nominal and uprated conditions.
4. This range is acceptable based on the simplified elastic-plastic analysis procedure given in NB-3228.5 of the ASME Code. An appropriate penalty factor is used in the fatigue evaluation for the inside surface of ASNs 13 (a fatigue evaluation is not performed for ASN 12). All other stress ranges at this ASN are less than the $3S_m$ limit of 69.90 ksi.

Bracketed [ ]^{a,c} information designates data that is Westinghouse Proprietary, as dicussed in Section 1.7 of this report.

Table 5.7-4 J-Nozzle-to-Feedring Weld Maximum Stress Intensity Ranges at Limiting Locations				
	Surface	Load Combination	Stress Intensity Range (ksi) (1)	ASME Code Limit
A	Outside	Loss of Load –Cold Shutdown	[ ]^{a,c}	69.9
B	Inside(2)	Loss of Load –Cold Shutdown	[ ]^{a,c}	69.9
C	Outside	Loss of Load –Cold Shutdown	[ ]^{a,c}	48.0

Notes:

1. Maximum ranges of stress intensity do not change for uprated conditions from values previously calculated.
2. Limiting for fatigue evaluation (additional fatigue strength reduction factor of 4 is used at inside surface of B to account for stress concentration at the weld root).

Bracketed [ ]^{a,c} information designates data that is Westinghouse Proprietary, as dicussed in Section 1.7 of this report.

Table 5.7-5

Summary of 7.4-Percent Uprate Power Operating Conditions[1]

Parameter	High T_{ave} Conditions	Low T_{ave} Conditions
P_{prim} (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.

5.8 Pressurizer Component Evaluations

To support the planned 7.4-percent Power Uprating, the pressurizer has been evaluated for operation at the upratedpower conditions. Any pressurizer design transients that were affected by the power uprate were addressed in the evaluations.

5.8.1 Pressurizer Evaluation

5.8.1.1 Introduction and Background

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 the 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 T_{steam} 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 plants have very similar pressurizer units. Both pressurizers were built to the same base design specification (Reference 1), and the units share the same original design basis analysis as documented in the plant-specific stress reports (References 2 and 3). The Performance Capability Working Group (PCWG) T_{hot} and T_{cold} parameters applicable for the Point Beach Replacement Steam Generator (RSG) 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 Analyses and Evaluations

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 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 condition related to the pressurizer have not changed from those applicable for the RSG Program.

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 4).

5.8.1.5 Results

Table 5.8-1 compares the operating conditions' change in temperature (ΔT) values for both the T_{hot} and T_{cold} 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 stress intensity range. For the spray nozzle, the uprate did not affect the maximum stress intensity range, either. However, since the maximum stress intensity range 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 the power uprate 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 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 4) limits (fatigue usage < 1.0) for both the spray and surge nozzles.

On this basis, the pressurizer is qualified for operation at the 7.4-percent uprated power level.

5.8.2 References

1. Equipment Specification 676440, *General Pressurizer Vessel Assembly*, E. H. Frye, Rev. 4, September 16, 1971.

2. P. P. DeRosa and W. R. Allen, *Pressurizer Stress Report, Wisconsin Electric Power Company, Point Beach Nuclear Plant Unit No. 2*, Sections 1 and 2, April 1975.

3. D. T. Entenmann and W. R. Allen, *Pressurizer Stress Report, Wisconsin Public Service Corporation, Kewaunee Power Station No. 1*, Sections 1 and 2, October 1974.

4. *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)						
		ΔT (°F)	RSG		7.4-% Uprate	
Component	Parameters	Original Design Basis	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

Table 5.8-2 Kewaunee Fatigue Usage Comparisons (ASME Code allowable < 1.0)		
Component	Revised RSG/7.4-% Uprate Fatigue Usage	Original Design Basis Fatigue Usage
Spray Nozzle	[ ]^{a,c}	[ ]^{a,c}
Surge Nozzle	[ ]^{a,c}	[ ]^{a,c}
Safety and Relief Nozzle	[ ]^{a,c}	[ ]^{a,c}
Lower Head	[ ]^{a,c}	[ ]^{a,c}
Support Skirt & Flange	[ ]^{a,c}	[ ]^{a,c}
Support Lug	[ ]^{a,c}	[ ]^{a,c}
Instrument Nozzle	[ ]^{a,c}	[ ]^{a,c}
Immersion Heater	[ ]^{a,c}	[ ]^{a,c}

Bracketed [ ]^{a,c} information designates data that is Westinghouse Proprietary, as discussed in Section 1.7 of this report.

Table 5.8-3 Maximum Primary-Plus-Secondary Stress Intensity Ranges			
Component	**SI Max. Range (ksi)**	**Uprated SI Max. Range (ksi)**	**ASME Code Limit (ksi)**
Surge Nozzle	[ ]^{a,c}	[ ]^{a,c}	58.2
Spray Nozzle	[ ]^{a,c}	[ ]^{a,c}	39.6$^{(1)}$

Note:

1. A simplified elastic-plastic analysis was performed in accordance with Section NB-3228.3 of the ASME Code, and the resultant stress intensity was less than the limit, and therefore, acceptable.

Bracketed [ ]^{a,c} information designates data that is Westinghouse Proprietary, as discussed in Section 1.7 of this report.

5.9 Nuclear Steam Supply System Auxiliary Equipment

5.9.1 Introduction and Background

This section evaluates the Kewaunee Nuclear Power Plant (KNPP) auxiliary tanks, heat exchangers, pumps, and valves on a system basis, for impact by the thermal transients and maximum operating temperatures, pressures, and flow rates resulting from the uprating conditions. The systems affected by the uprating are the Reactor Coolant System (RCS), Chemical and Volume Control System (CVCS), Safety Injection System (SIS), Residual Heat Removal System (RHRS), and the Component Cooling Water System (CCWS). The evaluation consists of a structural fatigue review and flow capacity review of the component pressure boundaries. The review does not include a structural evaluation or a performance/controllability evaluation of the sub-components for any of the main pressure retaining components listed in this report since they are not related to the main component pressure boundaries. These include: valve actuators, controllers, electronics, or pump motors that are covered by other in-plant monitoring, qualification, or test/inspection programs.

5.9.2 Input Parameters and Assumptions

The auxiliary system heat exchangers and tanks evaluated for uprate are listed in Tables 5.9-1 and 5.9-2. The auxiliary system pumps and valves are listed in Tables 5.9-3 and 5.9-4. The component design information is contained in the design documents listed in the tables. This information contained in these documents includes pressure and temperature design conditions as well as design transients applicable to each individual identifed component. The key input parameters used in the evaluation are the uprate parameters in Section 2 and the NSSS and auxiliary design transients. It is assumed that the listings in Tables 5.9-1 through 5.9-4 represent the actual hardware in the plant, and that any changes made to the auxiliary tanks, heat exchangers, pumps, and valves or operation of these components have been made by the plant, in accordance with the original technical and quality assurance requirements.

5.9.3 Description of Analyses and Evaluations

The design parameters were reviewed for the auxiliary tanks, heat exchanger, pumps, and valves. The specific criteria included design temperature, pressure, thermal transients, and flow rates. These parameters were compared to those used in the power uprate to determine if the design parameters still enveloped those for the uprating.

5.9.3.1 Auxiliary System Heat Exchangers

The NSSS auxiliary heat exchangers evaluated for the uprating conditions are listed in Table 5.9-1. The table includes design data that were used in the manufacture of each heat exchanger as well as the system(s) in which each heat exchanger is located. Several heat exchangers have two systems listed. This is because the tube-side and shell-side circulate water from two different systems. The specifications identify the applicable design transients and the data sheets identify the design temperature and pressures.

Based on the PCWG parameters for the uprating (Section 2), there is no impact on the auxiliary systems heat exchangers listed in Table 5.9-1. The operating temperature and pressure ranges for these vessels remain bounded by the original design parameters. The original design transients for the auxiliary equipment bound the transients associated with the uprating. The heat exchangers identified as having transients in the original design specifications are the regenerative, letdown, excess letdown, and residual heat removal (RHR) heat exchangers. All of these temperatures remain bounded by the original design conditions.

5.9.3.2 Auxiliary System Tanks

The only tanks for which transients are identified are the safety injection accumulators (see Table 5.9-2). The operating temperatures and pressures for these vessels remain within the design basis for these tanks, and the safety injection accumulators remain bounded by the original design transients. As a result, none of the auxiliary tanks are impacted by the uprating conditions.

Note that the pressurizer relief tank sparger is not included in this evaluation because it is not impacted by the uprating.

5.9.3.3 Auxiliary System Pumps

The NSSS auxiliary pumps evaluated for the uprating conditions are listed in Table 5.9-3. The table includes design data that were used in the manufacture of each pump. There is no impact on the auxiliary system pumps listed in Table 5.9-3 as a result of the uprating. The operating temperature and pressure ranges for these pumps remain bounded by the original design parameters. The original design transients for the auxiliary equipment bound the transients associated with the uprating.

5.9.3.4 Auxiliary System Valves

The NSSS auxiliary system valves that were evaluated for the uprating conditions are listed in Table 5.9-4. Table 5.9-4 also provides design data that were used in the manufacture of auxiliary valve. In addition, the table also lists the system(s) in which each valve is located. The specifications identify the applicable design transients, and the data sheets identify the design temperature and pressures.

There is no impact upon the auxiliary systems valves listed in Table 5.9-4 as a result of the uprating. The operating temperature and pressure ranges for the valves remain bounded by the original design parameters. The original design transients for the auxiliary equipment bound the transients associated with the power uprating.

5.9.4 Acceptance Criteria

To demonstrate equipment qualification, the maximum system operating temperatures, pressures, flow rates, and transients for the Uprating Program must be bounded by or equal to the original system design conditions as well as those applicable to the replacement steam generator (RSG).

5.9.5 Results

A comparison of the power uprate conditions shows that all maximum operating temperatures and pressures for systems evaluated are bounded by the existing design basis. Since all tanks, heat exchangers, pumps, and valves were designed and manufactured consistent with the system design and applicable codes and standards, all of the NSSS tanks, heat exchangers, pumps, and valves are acceptable for the maximum system operating temperatures and pressures resulting from the uprating. The auxiliary equipment and NSSS thermal transients resulting from the uprating are bounded by the original KNPP design parameters. Therefore, the auxiliary tanks, heat exchangers, pumps, and valves remain acceptable for the thermal transients resulting from the uprating. This is also applicable to any equipment for the above identifed systems that may have been changed or replaced in accordance with the original technical and quality assurance requirements.

5.9.6 Conclusions

The KNPP auxiliary tanks, heat exchangers, pumps, and valves are acceptable for the uprating conditions since there is no change to the auxiliary systems operating conditions identified as a consequence of the uprating.

The results for the uprating are consistent with, and continue to comply with, the current KNPP licensing basis/acceptance requirements. The original NSSS design parameters (for example, maximum and minimum temperatures) and the auxiliary design transients remain bounding for the conditions associated with the uprated power level.

Table 5.9-1 Kewaunee Auxiliary Heat Exchangers				
Component Supplier/System	**Spin/Drawing No.**	**Westinghouse Data Sheet**	**Equipment Specification**	**Purchase Order Number**
Regenerative Hx Joseph Oat./CVCS	WPS-CSAHRG 4968	AH-RG501	G-676454	546-CRH-98919
Residual Hx Joseph Oat./RHR	WPS-ACAHRS 4927	AH-RS547	G-676454	546-Z-98917
Seal Water Hx Atlas Ind./CVCS	WPS-CSAHSW D-1259	AH-SW508	G-676454	546-Z-70135
Excess Letdown Hx Sentry Equip. Corp./CVCS	WPS-CSAHEL P6703-4-1	AH-EL513	G-676454	546-Z-66179
Letdown Hx Atlas Ind./CVCS	WPS-CSAHLD D-2063	AH-LD549	G-676454	546-CRH-98915
CC Water Hx Eng. & Fab./CCW	WPS-ACAHCC CD-15748	AH-CC550	G-676454	546-CRH-98904

Table 5.9-2 Kewaunee Tanks		
Component/System	**Drawing/ESPEC**	**Rev.**
Component Cooling Surge Tank/CVCS	110E062	4
Boric Acid Tank/CVCS	695J523	3
Boric Acid Batching Tank/CVCS	110E084	2
Concentrates Holding Tank/CVCS	685J265	2
Chemical Mixing Tank/CVCS	5648D86	1
Holdup Tank/CVCS	D-69-518	6
Monitor Tank/CVCS	685J558	1
Resin Fill Tank/CVCS	617F668	1
VCT/CVCS	685J501	1
Boric Acid Filter/CVCS	51048	C
Concentrates Filter/CVCS	51044	C
Ion Exchange Filter/CVCS	51035	N/A
Seal Water Injection Filter/CVCS	D444-2008	0
Seal Water Return Filter/CVCS	51048	C
PRT/RCS	685J487	4
SIS Accumulator/SIS	47045	C

Table 5.9-3

Kewaunee Auxiliary Pumps

Spin Number	Description/System	Equipment Specification No./Rev	Vendor
CSAPBA	Boric Acid Transfer Pump/CVCS	103317/2	Goulds
ACAPCC	Component Cooling Water Pump/CCW	70491/2	I.D.P (I-R)
CSAPHT	Concentrate Holdup Tank Pump/CVCS	103322/2	Chempump
CSAPGS	Gas Stripper Feed Pump/CVCS	103322/1	Chempump
CSAPRE	Holdup Tank Recirc. Pump/CVCS	66168/1	I.D.P (I-R)
CSAPMT	Monitor Tank Pump	103317/2	Goulds
CSAPCH	Positive Displacement Charging Pump/CVCS	65265/3	Ajax Iron Works
ACAPRH	Residual Heat Removal Pump/RHR	86098/2	Byron Jackson
SIAPSI	Safety Injection Pump/SIS	103319/2	Bingham

Table 5.9-4

Kewaunee Auxiliary Valves

Valve ID																		
Size	Mode	CT	Special	Location	Sys Code	Qty Req	E-Spec PR-Gen	RV	SPS	RV	Drawing Number	RV	ST TR	Vendor	PO Number	C/N	C/N Qty	Spare Spin
3/8	T 58				RCS SIS	2 2	677264	0	10	3	9642			Kerotest	106765	002 030	4 2	GAVBMN-C
				No. Required 4			No. Assigned 6											
3/8	TA58DL			8032	RCS	1	676270	01	001		CP1-1-51			Mason Neil	130000	000	1	GAVBAO-T
				No. Required 1			No. Assigned 1											
3/8	TA58RL			8025 8026 9159A 9159B 9999A 9999B 9999C 9999D	RCS RCS WDS WDS SPAR SPAR SPAR SPAR	1 1 1 1 1 1 1 1	676270		2		CP1-1-52			Mason Neil	130000	000 003	6 2	GAVBAO-A
				No. Required 8			No. Assigned 8											
3/4	C 58				ACS CVCS RCS SIS	1 5 1 2	677264	0	5	2	C-464528			Edward	086071	010 042	8 2	GAVBMN-E
				No. Required 8			No. Assigned 10											
3/4	C 58A				CVCS	3	677264	0	6	1	C-465268			Edward	086071	010	3	GAVBMN-F
				No. Required 3			No. Assigned 3											

Table 5.9-4 (Cont.)

Kewaunee Auxiliary Valves

Valve ID																		
Size	Mode	CT	Special	Location	Sys Code	Qty Req	E-Spec PR-Gen	RV	SPS	RV	Drawing Number	RV	ST TR	Vendor	PO Number	C/N	C/N Qty	Spare Spin
3/4	DA42R			8027 8028 8149A 8149B	RCS RCS CVCS CVCS	1 1 1 1	676281	01	6		WAPD-CV-SS-2R	0		Grinnell	086075	017	4	GAVBAO-F
				No. Required 4			No. Assigned 4											
3/4	IA58RE		8145	8145	CVCS	1	678270	01	204		CP1-16-55RG A-842			Mason Neil	091713	003	1	GAVBAO-C
				No. Required 1			No. Assigned 1											
3/4	1A58RE		8824A	8824A	SIS	1	676270	01	205		CP1-18-55RQ A-842			Mason Neil	091713	003	1	GAVBAO-C
				No. Required 1			No. Assigned 1											
3/4	1A58RE		8824B	8824A	SIS	1	676270	01	205		CP1-18-55RQ A-842			Mason Neil	091713	003	1	GAVBAO-C
				No. Required 1			No. Assigned 1											
3/4	1A58RE		8825A	8825A	SIS	1	676270	01	205		CP1-18-55RG A-842			Mason Neil	091713	003	1	GAVBAO-C
				No. Required 1			No. Assigned 1											
3/4	1A58RE		8825B	8825B	SIS	1	676270	01	205		CP1-18-55RQ A-842			Mason Neil	091713	003	1	GAVBAO-C
				No. Required 1			No. Assigned 1											
3/4	PR52N	P		0441	RCS	1	677430	00	1		41900093		0	Veriflo	130050	000	1	GAVBRA-D
				No. Required 1			No. Assigned 1											

Table 5.9-4 (Cont.)

Kewaunee Auxiliary Valves

Valve ID		CT	Special	Location	Sys Code	Qty Req	E-Spec PR-Gen	RV	SPS	RV	Drawing Number	RV	ST TR	Vendor	PO Number	C/N	C/N Qty	Spare Spin
Size	Mode																	
3/4	RA52RDB		191	0191	CVCS	1	676270	01	1									GAVBAO-D
				No. Required 1			No. Assigned 1											
3/4	RV32SW		9420	9420	SPAR	1	676257	01	1		H-50911			Crosby	106776	001	1	GAVBSR-A
				No. Required 1			No. Assigned 0											
3/4	RV32SW		9421	9421	SPAR	1	676257	01	1		H-50911			Crosby	106776	001	1	GAVBSR-A
				No. Required 1			No. Assigned 1											
3/4	RV32SW		9423	9423	SPAR	1	676257	01	1		H-50911			Crosby	106776	001	1	GAVBSR-A
				No. Required 1			No. Assigned 1											
3/4	RV32SW		9424A	9424A	SPAR	1	676257	01	1		H-50911			Crosby	106776	001	1	GAVBSR-A
				No. Required 1			No. Assigned 1											
3/4	RV32SW		9424B	9424B	SPAR	1	676257	01	1		H-50911			Crosby	106776	001	1	GAVBSR-A
				No. Required 1			No. Assigned 1											
3/4	RV32SW		9426	9426	SPAR	1	676257	01	1		H-50911			Crosby	106776	001	1	GAVBSR-A
				No. Required 1			No. Assigned 1											
3/4	RV32SW		9429	9429	SPAR	1	676257	01	1		H-50911			Crosby	106776	001	1	GAVBSR-A
				No. Required 1			No. Assigned 1											
3/4	RV32SW		9430	9430	SPAR	1	676257	01	1		H-50911			Crosby	106776	001	1	GAVBSR-A
				No. Required 1			No. Assigned 1											
3/4	RV38SW		9427A	9427A	SPAR	1	676257	01	002		H-50914			Crosby	106776	072	1	GAVBSR-B
				No. Required 1			No. Assigned 1											

Table 5.9-4 (Cont.)

Kewaunee Auxiliary Valves

Valve ID																		
Size	Mode	CT	Special	Location	Sys Code	Qty Req	E-Spec PR-Gen	RV	SPS	RV	Drawing Number	RV	ST TR	Vendor	PO Number	C/N	C/N Qty	Spare Spin
3/4	RV38SW		9427B	9427B	SPAR	1	676257	01	002		H-50914			Crosby	106776	072	1	GAVBSR-A
				No. Required 1			No. Assigned 1											
3/4	RV58SW		8121A	8121A	CVCS	1	676257	01	6		H-50913-1			T Crosby	106776	001 061 061	1 –1 1	GAVBSR-D
				No. Required 1			No. Assigned 1											
3/4	RV58SW		8121A	8121A	CVCS	1	676257	01	6		H-50913-1			T Crosby	106776	001 061 061	1 –1 1	GAVBSR-D
				No. Required 1			No. Assigned 1											
3/4	RV58SW		8121C	8121C	CVCS	1	676257	01	6		H-50913-1			T Crosby	106776	001 061 061	1 –1 1	GAVBSR-D
				No. Required 1			No. Assigned 1											
3/4	T 36				ACS	50	677264	0	3	1	0-464531			Edward	086071	010 021	70 5	GAVBMN-G
3/4	T 58				ACS CVCS RCS SIS	13 40 33 18	677264	0	7	2	0-464530			Edward	086071	010	220	GAVBMN-H
3/4	TS42S			8104	CVCS	1			2	2	HV-168-727			ASCO	077745	007	1	GAVBSO-D
				No. Required 1			No. Assigned 1											
3/8	X 32D				CVCS	1	676281	2	6	3	WREF-2-CS			Grinnell	086075	011	1	GAVBDI-A
				No. Required 1			No. Assigned 1											

Table 5.9-4 (Cont.)

Kewaunee Auxiliary Valves

Valve ID		CT	Special	Location	Sys Code	Qty Req	E-Spec PR-Gen	RV	SPS	RV	Drawing Number	RV	ST TR	Vendor	PO Number	C/N	C/N Qty	Spare Spin
Size	Mode																	
3/8	X 42D				CVCS RCS	78 1	676281	2	5	3	WREF-2-SS			Grinnell Grinnell	065191 086075	047 017 052	10 84 65	GAVBDI-E
				No. Required 78			No. Assigned 159											
3/8	X 58N				CVCS RCS	4 2	677264	0	8	3	D-465285			Edward	086071	010	6	GAVBMNBB
				No. Required 6			No. Assigned 6											
1	C 58				CVCS	14	677264	0	5	2	C-464947			Edward	086071	010	18	GAVBMN-L
				No. Required 14			No. Assigned 8											
1	IA38RES		8820	8820	SIS	1	676270	01	215		CP1-18-55RG A-845			Mason Neil	091713	003 021 021	1 –1 1	GAVBAO-I
				No. Required 1			No. Assigned 1											
1	IA56RE		8823A	8823A	SIS	1	676270	01	209		CP1-18-55RG A-842			Mason Neil	091713	003	1	GAVBAO-E
				No. Required 1			No. Assigned 1											
1	IA56RE		8823B	8823B	SIS	1	676270	01	209		CP1-18-55RG A-842			Mason Neil	091713	003	1	GAVBAO-E
				No. Required 1			No. Assigned 1											
1	IA56RES		8821A	8821A	SIS	1	676270	01	210		CP1-18-55RG A-845			Mason Neil	091713	003 021 021	1 –1 1	GAVBAO-K
				No. Required 1			No. Assigned 1											

Table 5.9-4 (Cont.)

Kewaunee Auxiliary Valves

Valve ID		CT	Special	Location	Sys	Qty	E-Spec	RV	SPS	RV	Drawing	RV	ST	Vendor	PO	C/N	C/N	Spare Spin
Size	Mode				Code	Req	PR-Gen				Number		TR		Number		Qty	
1	IA56RES		8821B	8821B	SIS	1	676270	01	210		CP1-18-55RG A-845			Mason Neil	091713	003 021 021	1 –1 1	GAVBAO-K
				No. Required 1			No. Assigned 1											
1	IA58RF		8144	8144	CVCS	1	676270	01	210		CP1-18-55RG A-845			Mason Neil	091713	003	1	GAVBAO-L
				No. Required 1			No. Assigned 1											
1	IA58RF		8822A	8822A	SIS	1	676270	01	211		CP1-18-55RG A-842			Mason Neil	091713	003	1	GAVBAO-L
				No. Required 1			No. Assigned 1											
1	IA58RF		8822B	8822B	SIS	1	676270	01	211		CP1-18-55RG A-842			Mason Neil	091713	003	1	GAVBAO-L
				No. Required 1			No. Assigned 1											
1	PR52H	P		0113	CVCS	1	677430	00	3		41900091	0		Veriflo	130050	000	1	GAVBRA-B
				No. Required 1			No. Assigned 1											
1	PR52N	P		0114	CVCS	1	677430	00	2		41900094	0		Veriflo	130050	000	1	GAVBRA-A
				No. Required 1			No. Assigned 1											
1	PR58N	P		0944	SIS	1	677430	00	4		41900095	0		Veriflo	130050	000	1	GAVBRA-C
				No. Required 1			No. Assigned 1											
1	RA36RDS	H	945	0945	SIS	1	676270	01	214		CP1-18-55RG A-845			Mason Neil	091713	003 021 021	1 –1 1	GAVBAO-O
				No. Required 1			No. Assigned 1											

Table 5.9-4 (Cont.)

Kewaunee Auxiliary Valves

Valve ID Size	Valve ID Mode	CT	Special	Location	Sys Code	Qty Req	E-Spec PR-Gen	RV	SPS	RV	Drawing Number	RV	ST TR	Vendor	PO Number	C/N	C/N Qty	Spare Spin
1	RA42DO	F	110A	0110A	CVCS	1	676270	01	208		CP1-18-65RD A-845			Mason Neil	091713	003 021 021	1 –1 1	GAVBAO-P
			No. Required 1				No. Assigned 1											
1	RA58RF	H	123	0135	CVCS	1	676270	01	212									GAVBAO-Y
			No. Required 1				No. Assigned 0											
1	RV32ES		8125	8125	CVCS	1	676257	01	10		H-51345			Crosby	106776	001 109 109	1 –1 1	GAVBSR-G
			No. Required 1				No. Assigned 1											
1	RV32EW		9431A	9431A	SPAR	1	676257	01	9		H-51031			Crosby	106776	001	1	GAVBSR-J
			No. Required 1				No. Assigned 1											
1	RV32EW		9431A	9431A	SPAR	1	676257	01	9		H-51031			Crosby	106776	001	1	GAVBSR-J
			No. Required 1				No. Assigned 1											
1	RV52ENS		9151	9151	WDS	1	676257	01	27		H-51599			Crosby	106776	001	1	GAVBSR-K
			No. Required 1				No. Assigned 1											
1	RV56DGS		8830A	8830A	SIS	1	676257	01	11		H-51045			Crosby	106776	001 027 027	1 –1 1	GAVBSR-L
			No. Required 1				No. Assigned 1											
1	RV56DGS		8830B	8830B	SIS	1	676257	01	11		H-51045			Crosby	106776	001 027 027	1 –1 1	GAVBSR-L
			No. Required 1				No. Assigned 1											

Table 5.9-4 (Cont.)

Kewaunee Auxiliary Valves

Valve ID																		
Size	Mode	CT	Special	Location	Sys Code	Qty Req	E-Spec PR-Gen	RV	SPS	RV	Drawing Number	RV	ST TR	Vendor	PO Number	C/N	C/N Qty	Spare Spin
1	T 36				ACS	9	677264	0	3	1	D-464949			Edward	086071	010 021	8 1	GAVBMN-M
				No. Required 9			No. Assigned 9											
1	T 58				SIS	2	677264	0	7	2	D-464948			Edward	086071	010 142	2 4	GAVBMN-O
				No. Required 2			No. Assigned 6											
1	WA42RD		8146	8146	CVCS	1	676270	01	207		CP1-18-69RC A—8429			Mason Neil	091713	007	1	GAVBAO-J
				No. Required 1			No. Assigned 1											
1	X 420				CVCS	69	676281	2	5	3	WREF-3-SS			Grinnell	0860075	011	134	GAVBDI-J
				No. Required 129			No. Assigned 134											
1	X 59N				CVCS	2	677264	0	8	3	D-465284			Edward	086071	010	2	GAVBMNBC
				No. Required 2			No. Assigned 2											
1 1/2	C 38				ACS	2	677264	0	2	1	C-465272			Edward	086071	010	2	GAVBMNBF
				No. Required 2			No. Assigned 2											
1 1/2	IA38DFP		9450A	9450A	SPAR	1	676270	01	253		CP1-18-65RE A-845			Mason Neil	091713	003	1	GAVBAOAA
				No. Required 1			No. Assigned 1											
1 1/2	IA38DFP		9450B	9450B	SPAR	1	676270	01	253		CP1-18-65RE A-845			Mason Neil	091713	003	1	GAVBAOAA
				No. Required 1			No. Assigned 1											

Table 5.9-4 (Cont.)

Kewaunee Auxiliary Valves

Valve ID																		
Size	Mode	CT	Special	Location	Sys Code	Qty Req	E-Spec PR-Gen	RV	SPS	RV	Drawing Number	RV	ST TR	Vendor	PO Number	C/N	C/N Qty	Spare Spin
1 1/2	T 36				ACS	6	677264	0	3	1	D-467290			Edward	086071	010	6	GAVBMN-Q
				No. Required 6			No. Assigned 6											
2	C 36				ACS	1	677264	0	1	2	C-464526			Edward	086071	021	1	GAVBMN-R
				No. Required 1			No. Assigned 1											
2	C 58				CVS RCS SIS	25 1 3	677264	0	5	2	C-454529			Edward	085071	010 035	33 1	GAVBMN-S
				No. Required 29			No. Assigned 34											
2	C 58B				CVS	2	677264	0	6	1	C-465269			Edward	086071	010	2	GAVBMN-B
				No. Required 2			No. Assigned 2											
2	DA42R	R F F		001B 0110B 0110C 8029 8030 8031	NIRM CVCS CVCS RCS RCS RCS	1 1 1 1 1 1	676281		009		WAPD-CV- CS-GR	0		Grinnell	086075	017	7	GAVBAO-H
2	IA32RE	T	100	0100	CVCS	1	676270	01	226		CP1-18-71RD A-842			Mason Neil	091713	003 015 015	1 –1 1	
				No. Required 1			No. Assigned 1											
2	IA56RG		8147	8147	CVCS	1	676270	01	220		CP1-18-55RG A-842			Mason Neil	091713	003	1	GAVBAO-S
				No. Required 1			No. Assigned 1											

Table 5.9-4 (Cont.)

Kewaunee Auxiliary Valves

Valve ID		CT	Special	Location	Sys Code	Qty Req	E-Spec PR-Gen	RV	SPS	RV	Drawing Number	RV	ST TR	Vendor	PO Number	C/N	C/N Qty	Spare Spin
Size	Mode																	
2	IA58DG		8148A	8148A	CVCS	1	676270	01	227		CP1-18-65RE A-8456			Mason Neil	091713	007	1	GAVBAO-R
				No. Required 1			No. Assigned 1											
2	IA58RE	L	427	0427	RCS	1	676270	01	225		CP1-18-71RE A-8425			Mason Neil	091713	006 021 021	1 –1 1	GAVBAO-N
				No. Required 1			No. Assigned 1											
2	IA58RE	L	428	0428	RCS	1	676270	01	225		CP1-18-71RE A-8425			Mason Neil	091713	013 021 021	1 –1 1	GAVBAO-N
				No. Required 1			No. Assigned 1											
2	IA58RF		8142	8142	CVCS	1	676270	01	223		CP1-18-55RG A-847			Mason Neil	091713	007	1	GAVBAO-Y
				No. Required 1			No. Assigned 1											
2	IA58RF		8143	8143	CVCS	1	676270	01	223		CP1-18-55RG A-847			Mason Neil	091713	007	1	GAVBAO-Y
				No. Required 1			No. Assigned 1											
2	IA58RG		8140A	8140A	CVCS	1	676270	01	226		CP1-18-55RG A-842			Mason Neil	091713	003 008 008	1 –1 1	GAVBAO-X
				No. Required 1			No. Assigned 1											
2	IA58RG		8140B	8140B	CVCS	1	676270	01	226		CP1-18-55RG A-842			Mason Neil	091713	003 008 008	1 –1 1	GAVBAO-X
				No. Required 1			No. Assigned 1											

Table 5.9-4 (Cont.)

Kewaunee Auxiliary Valves

Valve ID																		
Size	Mode	CT	Special	Location	Sys Code	Qty Req	E-Spec PR-Gen	RV	SPS	RV	Drawing Number	RV	ST TR	Vendor	PO Number	C/N	C/N Qty	Spare Spin
2	IA58RG		8141	8141	CVCS	1	676270	01	226		CP1-18-55RG A-842			Mason Neil	091713	003 008 008	1 –1 1	GAVBAO-X
				No. Required 1			No. Assigned 1											
2	RA42RF	F	111	0111	CVCS	1	676270	01	217		CP1-18-55RG A-842			Mason Neil	091713	003 012 012	1 –1 1	GAVBAO-V
				No. Required 1			No. Assigned 1											
2	RA42RF	H	104	0104	CVCS	1	676270	01	218		CP1-18-55RG A-842			Mason Neil	091713	003	1	GAVBAO-V
				No. Required 1			No. Assigned 1											
2	RA42RF	H	105	0105	CVCS	1	676270	01	218		CP1-18-55RG A-842			Mason Neil	091713	003	1	GAVBAO-V
				No. Required 1			No. Assigned 1											
2	RA56DD	P	135	0135	CVCS	1	676270	01	218		CP1-18-6RD A-845			Mason Neil	091713	007 021 021	1 –1 1	GAVBAOAD
				No. Required 1			No. Assigned 1											
2	RA58DF	H	142	0142	CVCS	1	676270	01	252		CP1-18-65RD A-845			Mason Neil	091713	003	1	GAVBAOAE
				No. Required 1			No. Assigned 1											
2	RV52JWB		8116	8116	CVCS	1	676257	01	15		H-51382-2	0		Crosby	106776	001	1	GAVBSR-N
				No. Required 1			No. Assigned 1											

Table 5.9-4 (Cont.)

Kewaunee Auxiliary Valves

Valve ID Size	Valve ID Mode	CT	Special	Location	Sys Code	Qty Req	E-Spec PR-Gen	RV	SPS	RV	Drawing Number	RV	ST TR	Vendor	PO Number	C/N	C/N Qty	Spare Spin
2	RV52JWB		8119	8119	CVCS	1	676257	01	14		H-51382			Crosby	106776	001	1	GAVBSR-N
				No. Required 1			No. Assigned 1											
2	RV56JMB		8115	8115	CVCS	1	676257	01	17		H-51680-3			T Crosby	106776	001 045 045	1 –1 1	GAVBSR-P
				No. Required 1			No. Assigned 1											
2	RV56JWB		8705	8705	RHRS	1	676257	01	16		H-51680-1			Crosby	106776	001 109 109	1 –1 1	GAVBSR-M
				No. Required 1			No. Assigned 1											
2	T 36				ACS CVCS	11 1	677264	0	3	1	D-464533			Edward	086071	010 021	7 5	GAVBMN-U
				No. Required 12			No. Assigned 12											
2	T 58				ACS CVCS RCS SIS	4 17 17 17	677264	0	7	2	D-464532			Edward	086071	010 097 141	64 4 10	GAVBMN-V
				No. Required 55			No. Assigned 78											
2	TM58FN			8102 8103 8826A 8826B 9410	CVCS CVCS SIS SIS SPAR	1 1 1 1 1	676258		002		137 . 116-1			Velan	091710	008 022	6 –1	GAVBMO-H
				No. Required 5			No. Assigned 5											

Table 5.9-4 (Cont.)

Kewaunee Auxiliary Valves

Valve ID																		
Size	Mode	CT	Special	Location	Sys Code	Qty Req	E-Spec PR-Gen	RV	SPS	RV	Drawing Number	RV	ST TR	Vendor	PO Number	C/N	C/N Qty	Spare Spin
2	TM58FNH			8801A 8801B 8802A 8802B	SIS SIS SIS SIS	1 1 1 1	676258	01	2		137 . 116-1			Velan	091710	022	4	GAVBMO-Z
				No. Required 4			No. Assigned 4											
2	WA42DG	T	145	0145	CVCS	1	676270	01	232		CP2-8-53RD A-8459			Mason Neil	091713	007	1	GAVBAOAF
				No. Required 1			No. Assigned 1											
2	WA42RE	L	112A	0112A	CVCS	1	676270	01	233		CP2-8-54RD			Mason Neil	091713	003 007	1 –1	GAVB
				No. Required 1			No. Assigned 0											
2	WA52RO		112A	0112A	CVCS	1	676270	01	233									GAVBAOAC
				No. Required 1			No. Assigned 0											
2	WA52RG		8152	8152	CVCS	1	676270	01	233									GAVBADAC
				No. Required 1			No. Assigned 0											
2	X 32D				CVCS	1	676281	2	6	3	WREF-4-CS			Grinnell	086075	011 047	1 2	GAVBDI-K
				No. Required 1			No. Assigned 3											

Table 5.9-4 (Cont.)

Kewaunee Auxiliary Valves

Valve ID					Sys	Qty	E-Spec				Drawing		ST		PO		C/N	
Size	Mode	CT	Special	Location	Code	Req	PR-Gen	RV	SPS	RV	Number	RV	TR	Vendor	Number	C/N	Qty	Spare Spin
2	X 420			 8185A 8185B 8185C 8186A 8186B 8186C	CVCS RCS SIS CVCS CVCS CVCS CVCS CVCS CVCS	134 2 1 1 1 1 1 1 1	676281	01	5	3	WREF-4-SS			Grinnell	086076	011 082	144 5	GAVBDI-L
				No. Required 143			No. Assigned 149											
3	C 32			9454	SPAR	1	676241	01	4		73908-1			Velan	091709	023	1	GAVAMN-A
				No. Required 1			No. Assigned 1											
3	C 42			8196	CVCS CVCS	1 0	676241	01	33		D-48148			Darling Aloyco	103274 116879	008 013 035	1 –1 1	GAVAMN-P
				No. Required 1			No. Assigned 1											
3	G 32			9407A 9407B 9432 9434 9453	SPAR SPAR SPAR SPAR SPAR	1 1 1 1 1	676241	01	124		83309-1			Velan	091709	007 057 060	5 –1 1	GAVAMN-NB
				No. Required 5			No. Assigned 5											
3	G 58			8001A 8001B 8814A 8814B	RCS RCS SIS SIS	1 1 1 1	676241	01	132		88406			Velan	091709	023 063	4 1	GAVAMN-C
				No. Required 4			No. Assigned 5											

Table 5.9-4 (Cont.)

Kewaunee Auxiliary Valves

Valve ID					Sys	Qty	E-Spec				Drawing		ST		PO		C/N	
Size	Mode	CT	Special	Location	Code	Req	PR-Gen	RV	SPS	RV	Number	RV	TR	Vendor	Number	C/N	Qty	Spare Spln
3	GM32FB			9451	SPAR	1	676258	01	5		A-40063			Crane	070526	028	1	GAVAMO-N
				No. Required 1			No. Assigned 1											
3	GM42FB			8100A	CVCS	1	676258	01	6		E-47380			Aloyco	116880	008 015 015	1 –1 1	GAVAMO-A
				No. Required 1			No. Assigned 1											
3	GM42FBH			8100B	CVCS	1	676258	01			E-48022	0		Aloyco	116880	024	1	GAVAMO-B
				No. Required 1			No. Assigned 1											
3	GM58FN			8000A 8000B 8806A 8806B	RCS RCS SIS SIS	1 1 1 1	676268	01	7		88405-2			Velan	091710	008 022	2 2	GAVAMO-I
				No. Required 4			No. Assigned 4											
3	IA58RGP	P	430	0430	RCS	1	676270	01	239		CP1-18-55RG A-842			Mason Neil	091713	003	1	GAVAAO-A
				No. Required 1			No. Assigned 1											
3	IA58RGP	P	431C	0431C	RCS	1	676270	01	239		CP1-18-55RG A-842			Mason Neil	091713	003	1	GAVAAO-A
				No. Required 1			No. Assigned 1											
3	RA58RGA	P	431A	0431A	RCS	1	676270	01	241		CP1-18-56RD A-8430			Mason Neil	091713	003 051 051	1 –1 1	GAVAAO-E
				No. Required 1			No. Assigned 1											

Table 5.9-4 (Cont.)

Kewaunee Auxiliary Valves

Valve ID Size	Valve ID Mode	CT	Special	Location	Sys Code	Qty Req	E-Spec PR-Gen	RV	SPS	RV	Drawing Number	RV	ST TR	Vendor	PO Number	C/N	C/N Qty	Spare Spin
3	RA58RGA	P	431A	0431A	RCS	1	676270	01	241		CP1-18-56RO A-8430			Mason Neil	091713	003 051 051	1 –1 1	GAVAAO-E
				No. Required 1			No. Assigned 1											
3	RV32LNS		8124A	8124A	CVCS	1	676270	01	20		H-51036-2			Crosby	106776	001	1	GAVASR-A
				No. Required 1			No. Assigned 1											
3	RV32LNS		8124B	8124B	CVCS	1	676257	01	20		H-51036-2			Crosby	106776	001	1	GAVASR-A
				No. Required 1			No. Assigned 1											
3	RV32LNS		8124A	8124C	CVCS	1	676257	01	20		H-51036-2			Crosby	106776	001	1	GAVASR-A
				No. Required 1			No. Assigned 1											
3	RV32LW		941B	941B	SPAR	1	676257	01	18		H-51036			Crosby	106776	001	1	GAVASR-D
				No. Required 1			No. Assigned 1											
3	RV32LW		9428A	9428A	SPAR	1	676257	01	19		H-51036			Crosby	106776	001	1	GAVASR-D
				No. Required 1			No. Assigned 1											
3	RV32LW		9428B	9428B	SPAR	1	676257	01	19		H-51036			Crosby	106776	001	1	GAVASR-D
				No. Required 1			No. Assigned 1											
3	RV52LNS		8120	8120	CVCS	1	676257	01	21		H-51036-2			Crosby	106776	001 061 061	1 –1 1	GAVASR-B
				No. Required 1			No. Assigned 1											

Table 5.9-4 (Cont.)

Kewaunee Auxiliary Valves

Valve ID					Sys	Qty	E-Spec				Drawing		ST		PO		C/N	
Size	Mode	CT	Special	Location	Code	Req	PR-Gen	RV	SPS	RV	Number	RV	TR	Vendor	Number	C/N	Qty	Spare Spin
3	T 36			9406 9433 9452 9456A 9456B 9457A 9457B	SPAR SPAR SPAR SPAR SPAR SPAR SPAR	1 1 1 1 1 1 1	676241	01	144		73426			Velan	091709	023 063	7 1	GAVAMN-D
				No. Required 7			No. Assigned 8											
3	T 58			8166A 8166B 8166C 8168	CVCS CVCS CVCS CVCS	1 1 1 1	676241	01	145		13920			Velan Velan	067760 081709	125 140 041 056 099 102	4 –4 4 –4 4 1	GAVAMN-E
				No. Required 4			No. Assigned 5											
3	X 42D			8160 8161 8163 8165A 8165B 8165C 8171 8172 8173 8174 8175 8176 8177 8178 8179	CVCS CVCS CVCS CVCS CVCS CVCS CVCS CVCS CVCS CVCS CVCS CVCS CVCS CVCS CVCS	1 1 1 1 1 1 1 1 1 1 1 1 1 1 1	676281	2	5	3	WREF-5-SS			Grinnel	086075	011 035	34 1	GAVADI-A

Table 5.9-4 (Cont.)

Kewaunee Auxiliary Valves

Valve ID																		
Size	Mode	CT	Special	Location	Sys Code	Qty Req	E-Spec PR-Gen	RV	SPS	RV	Drawing Number	RV	ST TR	Vendor	PO Number	C/N	C/N Qty	Spare Spin
3	X 42D Continued			8187A 8187B 8187C 8190 8191 8192 8193 8194A 8194B 8195A 8195B 8156A 8156B	CVCS CVCS CVCS CVCS CVCS CVCS CVCS CVCS CVCS CVCS CVCS WDS WDS	1 1 1 1 1 1 1 1 1 1 1 1 1	676281	2	5	3	WREF-5-SS			Grinnel	086075	011 035	34 1	GAVADI-A
				No. Required 29			No. Assigned 35											
4	C 32			9404A 9404B	SPAR SPAR	1 1	676241	01	102		73908-2			Velan	091709	023	2	GAVAMN-F
				No. Required 2			No. Assigned 2											
4	C 42			8164 8170 8183	CVCS CVCS CVCS	1 1 1	676241	01	108		1026-3			Darling	103274	001 003	2 1	GAVAMN-G
				No. Required 3			No. Assigned 3											
4	C 58			8812A 8812B	SIS SIS	1 1	676241	01	111		78501			Velan	091709	023	2	GAVAMN-U
				No. Required 2			No. Assigned 2											
4	FA37RG	F	480	0480	SD	1	676270	01	1		CV4588RA EV991			Fisher Gov	103275	022	1	GAVAAO-D
				No. Required 1			No. Assigned 1											

Table 5.9-4 (Cont.)

Kewaunee Auxiliary Valves

Valve ID																		
Size	Mode	CT	Special	Location	Sys Code	Qty Req	E-Spec PR-Gen	RV	SPS	RV	Drawing Number	RV	ST TR	Vendor	PO Number	C/N	C/N Qty	Spare Spin
4	FA37RG	F	481	0480	SD	1	676270	01	1		CV4588RA EV991			Fisher Gov	103275	022	1	GAVAAO-D
				No. Required 1			No. Assigned 1											
4	G 32			9409 9415 9416	SPAR SPAR SPAR	1 1 1	676241	01	125		K-7358			Crane	103281	028 033 034 034	2 1 –2 2	GAVAMN-H
				No. Required 3			No. Assigned 3											
4	G 42			8169	CVCS CVCS	1 0	676241	01	42		00	0		Aloyco Darling	067765 103274	113 001 003 034	1 1 1 1	GAVAMN-I
				No. Required 1			No. Assigned 2											
4	G 58			8813A 8813B	SIS SIS	1 1	676241	01	134		88503			Velan	091709	023	2	GAVAMN-K
				No. Required 2			No. Assigned 2											
4	GM32FB			9401A 9401B 9402A 9402B	SPAR SPAR SPAR SPAR	1 1 1 1	676258	01	8		A-40007-B			Crane Crane	070526 116878	022 029 029 006 023	4 –4 4 4 –4	GAVAMO-D
				No. Required 4			No. Assigned 4											
4	GM42FB	L L		0112B 0112C	CVCS CVCS	1 1	6762158	01	9		15-1029-3			Darling	103251	006 020 020	2 2 –2	GAVAMO-C
				No. Required 2			No. Assigned 2											

Table 5.9-4 (Cont.)

Kewaunee Auxiliary Valves

Valve ID																		
Size	Mode	CT	Special	Location	Sys Code	Qty Req	E-Spec PR-Gen	RV	SPS	RV	Drawing Number	RV	ST TR	Vendor	PO Number	C/N	C/N Qty	Spare Spin
4	T 32			9417 9455A 9455B	SPAR SPAR SPAR	1 1 1	676241	01	29		73423	0		Velan Velan	067760 091709	085 058	1 2	GAVAMN-L
				No. Required 3			No. Assigned 3											
4	X 42D			8167 8180A 8180B 8180C 8181A 8181B 8181C 8182 8184	CVCS CVCS CVCS CVCS CVCS CVCS CVCS CVCS CVCS	1 1 1 1 1 1 1 1 1	676281	2	5	3	WREF-6-SS			Grinnell	086075	017	9	GAVADI-B
				No. Required 9			No. Assigned 9											
6	BA32D	T		0130	CVCS	1	676368	1	006	6	F-41278 H-11062			Contintal	103250	001 003 003 013 013	1 –1 1 1 –1	GAVABU-F
				No. Required 1			No. Assigned 1											
6	BA54R	H		0626	ACS	1	67636B	1	005	5	F-41300 H-11052			Contintal	103250	006	1	GAVABU-I
				No. Required 1			No. Assigned 1											

Table 5.9-4 (Cont.)

Kewaunee Auxiliary Valves

Valve ID																		
Size	Mode	CT	Special	Location	Sys Code	Qty Req	E-Spec PR-Gen	RV	SPS	RV	Drawing Number	RV	ST TR	Vendor	PO Number	C/N	C/N Qty	Spare Spin
6	C 58			8842A 8842B 8843A 8843B 8844A 8844B	SIS SIS SIS SIS SIS SIS	1 1 1 1 1 1	676241	01	113		78704			Velan	091709	023	6	GAVAMN-Q
				No. Required 6			No. Assigned 6											
6	G 32			9408A 9408B 9414 9419A 9419B	SPAR SPAR SPAR SPAR SPAR	1 1 1 1 1	676241		126		83306-2			Velan	091709	023 055	4 1	GAVAMN-R
				No. Required 5			No. Assigned 5											
6	G 54			8718	RHRS	1	676241	01	136		E-47385		0	Aloyco	116879	034	1	GAVAMN-M
				No. Required 1			No. Assigned 1											
6	GM42FB			8807A 8807B	SIS SIS	1 1	676258	01	13		1055-3			Darling	103251	001	2	GAVAMO-O
				No. Required 2			No. Assigned 2											
6	GM54FE			8816A 8816B	SIS SIS	1 1	676258	01	14		E-47671	0		Aloyco	116880	012 015 015	2 –2 2	GAVAMO-P
				No. Required 2			No. Assigned 2											
6	GM58FMH			8803A 8803B	SIS SIS	1 1	676258	01	505		88701-3			Velan	091710	008	2	GAVAMO-Q
				No. Required 2			No. Assigned 2											

Table 5.9-4 (Cont.)

Kewaunee Auxiliary Valves

Valve ID																		
Size	Mode	CT	Special	Location	Sys Code	Qty Req	E-Spec PR-Gen	RV	SPS	RV	Drawing Number	RV	ST TR	Vendor	PO Number	C/N	C/N Qty	Spare Spin
6	RV58LSB		8010A	8010A	RCS	1	676279	01	1									GAVASR-C
				No. Required 1			No. Assigned 0											
6	RV58LSB		8010B	8010B	RCS	1	676279	01	1									GAVASR-C
				No. Required 1			No. Assigned 0											
8	BA540	H H		0624 0625	ACS ACS	1 1	676360	1	005	F	F-41304 H-11051			Contintal	103250	001 003 003	2 –2 2	GAVABU-M
				No. Required 2			No. Assigned 2											
8	C 32			9403	SPAR	1	676241	01	104		K-7604			Crane	103281	007	1	GAVAMN-X
				No. Required 1			No. Assigned 1											
8	C 54			8710A 8710B 8712A 8712B	RHRS RHRS RHRS RHRS	1 1 1 1	676241	01	115		K-7422			Crane	103281	007 034 034	4 –4 4	GAVAMN-V
				No. Required 4			No. Assigned 4											
8	G 42			8815A 8815B	SIS SIS	1 1	676241	01	137		E-47388	0		Alovco	116879	005 029 029 076 076	2 –2 2 –2 2	GAVAMN-O
				No. Required 2			No. Assigned 2											

Table 5.9-4 (Cont.)

Kewaunee Auxiliary Valves

Valve ID		CT	Special	Location	Sys Code	Qty Req	E-Spec PR-Gen	RV	SPS	RV	Drawing Number	RV	ST TR	Vendor	PO Number	C/N	C/N Qty	Spare Spin
Size	Mode																	
8	G 54			8711A	RHRS	1	676241	01	138		K-7420			Crane	103281	020	7	GAVAMN-J
				8711B	RHRS	1										020	6	
				8713A	RHRS	1										033	–1	
				8713B	RHRS	1										034	–7	
				8714A	RHRS	1										034	7	
				8714B	RHRS	1										034	–5	
				8715A	RHRS	1										034	5	
				8715B	RHRS	1												
				8716A	RHRS	1												
				8716B	RHRS	1												
				8717A	RHRS	1												
				8717B	RHRS	1												
				No. Required 12			No. Assigned 12											
8	GM32FB			9400	SPAR	1	676258	01	16					Crane	070526	028	1	GAVAMO-R
																029	–1	
																029	1	
				No. Required 1			No. Assigned 1											
8	GM42FB			8809A	SIS	1	676256	01	17		15-1028-3			Darling	103251	001	2	GAVAMO-E
				8809B	SIS	1										006	1	
				8809C	SIS	1												
				No. Required 3			No. Assigned 3											
8	GM58SD			8701A	RHRS	1	676258	01	8		88802-4			Velan	091710	008	4	GAVAMO-M
				8701B	RHRS	1												
				8702A	RHRS	1												
				8702B	RHRS	1												
				No. Required 4			No. Assigned 4											
8	RA36RG	F	468A	0468A	SD	1	676270	01	2									GAVAAO-H
				No. Required 1			No. Assigned 0											

Table 5.9-4 (Cont.)

Kewaunee Auxiliary Valves

Valve ID																		
Size	Mode	CT	Special	Location	Sys Code	Qty Req	E-Spec PR-Gen	RV	SPS	RV	Drawing Number	RV	ST TR	Vendor	PO Number	C/N	C/N Qty	Spare Spin
8	RA36RG	F	468B	0468B	SD	1	678270	01	2									GAVAAO-H
				No. Required 1			No. Assigned 0											
8	RA36RG	F	468C	0468C	SD	1	876270	01	2									GAVAAO-H
				No. Required 1			No. Assigned 0											
8	RA36RG	F	478A	0478A	SD	1	676270	01	2		M-142249R3 B-1422	0		Copes Vulc	091676	003 015 015	1 –1 1	GAVAAO-H
				No. Required 1			No. Assigned 1											
8	RA36RG	F	478B	0478B	SD	1	676270	01	2		M-142249R3 B-1422	0		Copes Vulc	091676	003 015 015	1 1 –1	GAVAAO-H
				No. Required 1			No. Assigned 1											
8	RA36RG	F	478C	0478C	SD	1	676270	01	2		M-142249R3 B-1422	0		Copes Vulc	091676	003 015 015	1 1 –1	GAVAAO-H
				No. Required 1			No. Assigned 1											
8	RA36RG	F	484A	0484A	SD	1	676270	01	1		M-142249R3 B-1422	0		Copes Vluc	091676	003 015 015	1 –1 1	GAVAAO-H
				No. Required 1			No. Assigned 1											
8	RA36RG	F	484B	0484B	SD	1	676270	01	2		M-142249R3 B-1422	0		Copes Vulc	091676	003 015 015	1 –1 1	GAVAAO-H
				No. Required 1			No. Assigned 1											

Table 5.9-4 (Cont.)

Kewaunee Auxiliary Valves

Valve ID																		
Size	Mode	CT	Special	Location	Sys Code	Qty Req	E-Spec PR-Gen	RV	SPS	RV	Drawing Number	RV	ST TR	Vendor	PO Number	C/N	C/N Qty	Spare Spin
8	RA36RG	F	484C	0484C	SD	1	676270	01	2		M-142249R3 B-1422	0		Copes Vulc	091676	003 015 015	1 1 –1	GAVAAO-H
				No. Required 1			No. Assigned 1											
8	RA36RG	F	484D	0484D	SD	1	676270	01	1		M-142249R3 B-1422	0		Copes Vulc	091676	003	1	GAVAAO-H
				No. Required 1			No. Assigned 1											
8	RA36RG	F	484E	0484E	SD	1	676270	01	1		M-142249R3 B-1422	0		Copes Vulc	091676	003	1	GAVAAO-H
				No. Required 1			No. Assigned 1											
8	RA36RG	F	484F	0484F	SD	1	676270	01	1		M-142249R3 B-1422	0		Copes Vulc	091676	003	1	GAVAAO-H
				No. Required 1			No. Assigned 1											
10	C 32			9441A 9441B	SPAR SPAR	1	676241	00	107		73908-4			Velan Velan	067760 091709	093 023	1 1	GAVAMN-Y
				No. Required 2			No. Assigned 2											
10	C 54			8811A 8811B	SIS SIS	1 1	676241	01	117		D-47413	0		Aloyco	116879	005	2	GAVAMN-S
				No. Required 2			No. Assigned 2											
10	G 32			9412A 9442A 9442B	SPAR SPAR SPAR	1	676241	01	128		83309-4			Velan	091709	023	4	GAVAMN-T
				No. Required 4			No. Assigned 4											

Table 5.9-4 (Cont.)

Kewaunee Auxiliary Valves

Valve ID																		
Size	Mode	CT	Special	Location	Sys Code	Qty Req	E-Spec PR-Gen	RV	SPS	RV	Drawing Number	RV	ST TR	Vendor	PO Number	C/N	C/N Qty	Spare Spin
10	GM32SB			9411A 9411B	SPAR SPAR	1 1	676255	01	22		K-7629			Crane	116878	006	2	GAVAMO-F
				No. Required 2			No. Assigned 2											
10	GM54SE			8810A 8810B	SIS SIS	1 1	676258	01	26		K-7633			Crane	116878	006 025 025	2 –2 2	GAVAMO-Q
				No. Required 2			No. Assigned 2											
10	VM585D			8703	RHRS	1	676258	01	525		93-13021			Darling	103251	006	1	GAVAMO-H
				No. Required 1			No. Assigned 1											
10	X 32B			9413A 9413B	SPAR SPAR	1 1	787379	1	003	5	F-41512 H-11061			Contintal	103250	001	2	GAVABU-O
				No. Required 2			No. Assigned 2											
12	C 48Z			8840A 8840B 8841A 8841B	SIS SIS SIS SIS	1 1 1 1	676241	01	123		11542			Darling	091661	002 009	6 –2	GAVAMN-W
				No. Required 4			No. Assigned 4											
12	G 32			9440A 9440B 9443A 9443B 9444A 9444B	SPAR SPAR SPAR SPAR SPAR SPAR	1 1 1 1 1 1	676241	1	129		K-7359-5			Crane	103281	007 020	1 5	GAVAMNBD
				No. Required 6			No. Assigned 8											

Table 5.9-4 (Cont.)

Kewaunee Auxiliary Valves

Valve ID																		
Size	Mode	CT	Special	Location	Sys Code	Qty Req	E-Spec PR-Gen	RV	SPS	RV	Drawing Number	RV	ST TR	Vendor	PO Number	C/N	C/N Qty	Spare Spin
12	GM42F			8808A 8808B	SIS SIS	1 1	676258	01	551		E-47376	0		Aloyco	116880	003 015 015	2 –2 2	GAVAMO-L
				No. Required 2			No. Assigned 2											
12	GM54SE			8804A 8804B 8805A 8805B	SIS SIS SIS SIS	1 1 1 1	676258	01	33		K-7635			Crane	116878	006 025 025	4 –4 4	GAVAMO-J
				No. Required 4			No. Assigned 4											
12	GM58FJH		8800A	8800A	SIS	1	676258	1	517	0	88907			T Velan				GAVAMO-K
				No. Required 1			No. Assigned 0											
12	GM58FJH		8800B	8800B	SIS	1	676258	1	517	0	88907			T Velan				GAVAMO-K
				No. Required 1			No. Assigned 0											
16	FA37RG	F	466	0466	FW	1	676270	01	1		M-137874R4 M-1441	0		Copes Vulc	091698	004	1	GAVAAO-M
				No. Required 1			No. Assigned 0											
16	FA37RG	F	476	0476	FW	1	676270	01	1		M-137874R4 M-1441	0		Copes Vluc	091699	004	1	GAVAAO-M
				No. Required 1			No. Assigned 1											

6.0 NUCLEAR STEAM SUPPLY SYSTEM ACCIDENT ANALYSES

This section provides the results of the analyses and/or evaluations that were performed for the Nuclear Steam Supply System (NSSS) accident analyses in support of the Power Uprate Program. The accident analysis areas addressed in this section include:

- Loss-of-coolant accidents (LOCA)
- Non-LOCA events, including Anticipated Transients without Scram (ATWS)
- Steam generator tube rupture (SGTR) transient
- Containment integrity
- Main steamline break (MSLB) consequences
- LOCA hydraulic forces
- Radiological consequences (doses)
- Initial condition uncertainties and protection setpoints

The detailed results and conclusions of each analysis are presented within each subsection.

6.1 Loss-of-Coolant Accidents

6.1.1 Best-Estimate Large-Break Loss-of-Coolant Accident

6.1.1.1 Introduction and Background

The best-estimate large-break loss-of-coolant accident (BELBLOCA) analysis completed as part of the 422V+ fuel transition was performed consistent with the uprated core power level of 1772 MWt. The plant operating ranges that are supported by the analysis were confirmed by KNPP in Reference 2. The following is a summary of the BELBLOCA analysis performed for the fuel transition and stretch uprate.

The Westinghouse best-estimate LOCA (BELOCA) analysis methodology for three- and four-loop Westinghouse pressurized water reactors (PWRs) with cold-leg Emergency Core Cooling System (ECCS) injection is documented in WCAP-12945-P-A (Reference 3) and was approved by Reference 4 The methodology was extended to Westinghouse two-loop PWRs having residual heat removal (RHR) injection to the upper plenum in WCAP-14449-P-A, Revision 1 (Reference 5), and was approved by Reference 6. Westinghouse applied this approved methodology to analyze the large-break LOCA (LBLOCA) at the uprated core power level of 1772 MWt for the KNPP 422V+ fuel transition and stretch uprate.

For the best-estimate upper plenum injection (BE-UPI) evaluation model (EM), the Reference 4 *Safety Evaluation Report* (SER) requirements for three- and four-loop analyses apply. Several additional SER requirements specific to the BE-UPI EM are specified in Reference 6. These additional SER requirements require that the plant conditions analyzed in WCOBRA/TRAC fall within the range of test simulations used to assess the phenomena unique to UPI plants. All of the Reference 4 SER requirements have been met, and Table 6.1-1 demonstrates that the KNPP analysis parameters fall within the range of the UPI test conditions as required by the BE-UPI EM SER.

6.1.1.2 Description of Analyses and Evaluations

Transition cores containing both Westinghouse 422V+ fuel and partially depleted Framatome/ANP fuel were analyzed to determine the effects of hydraulic and design differences of the two fuel types. The transition core analysis demonstrated that the best-estimate analysis of a full 422V+ core bounds the results for transition cycles, as discussed in Section 5.2.1.4 of the KNPP RTSR (Reference 1).

Unlike previous Appendix K analyses for KNPP that conservatively bound the majority of uncertainty parameters, the BE-UPI analysis performed for the fuel transition and stretch uprate provides for realistic treatment of the uncertainties that have generally been bounded in past analyses. The calorimetric uncertainty associated with core power is one such uncertainty parameter. The BE-UPI analysis for KNPP models a nominal core power of 1772 MWt, and the calorimetric uncertainty is ranged ± 0.6 percent.

6.1.1.3 Acceptance Criteria and Results

Table 6.1-2 presents the BE-UPI LBLOCA analysis results. All acceptance criteria are met.

6.1.1.4 Conclusions

The BE-UPI LBLOCA analysis supports combinations of power and uncertainty such that the total core power inclusive of uncertainty does not exceed 1783 MWt and the nominal core power does not exceed 1772 MWt.

6.1.2 Small-Break Loss-of-Coolant Accident

6.1.2.1 Introduction and Background

The small-break LOCA (SBLOCA) analysis completed as part of the 422V+ fuel transition (Reference 1) was performed consistent with the uprated core power level of 1772 MWt and other plant operating conditions for the stretch uprate program. The plant operating ranges that are supported by the analysis were confirmed by the KNPP in Reference 2. The following is a summary of the SBLOCA analysis performed for the fuel transition and stretch uprate.

Westinghouse analyzed the SBLOCA using the NOTRUMP code (Reference 7) and the SBLOCTA code, a small-break-specific version of the LOCTA-IV code (Reference 8). The approved methodology used for the analysis is documented in WCAP-10054-P-A and WCAP-10054-P-A Addendum 2 (References 9 and 10), and extended in References 11 and 12.

6.1.2.2 Description of Analyses and Evaluations

The complete small-break spectrum of 2-, 3-, 4-, and 6-inch breaks was analyzed for a full core of Westinghouse 422V+ fuel at the uprated core power level of 1772 MWt, plus a 0.6-percent allowance for calorimetric uncertainty. Hydraulic mismatch is not a factor for SBLOCA as was demonstrated in WCAP-10444-P-A (Reference 13); therefore, the analysis for a full 422V+ core at the uprated power level applies for the transition cycles with partially depleted Framatome/ANP fuel in the core.

The results of the SBLOCA break spectrum at the high assumed T_{avg} value of 583.0°F, plus a 3-inch break confirmatory case at the low assumed T_{avg} value of 546.3°F, are presented in Table 6.1-3. The limiting break was found to be the 3-inch break initiated at the high assumed T_{avg}. The peak cladding temperature was 1030°F, local cladding oxidation was less than 17 percent, and core-wide oxidation was less than 1 percent.

6.1.2.3 Acceptance Criteria and Results

Table 6.1-3 presents the SBLOCA analysis results. All acceptance criteria are met.

6.1.2.4 Conclusions

The SBLOCA analysis supports combinations of core power and uncertainty such that the total core power inclusive of uncertainty does not exceed 1783 MWt.

6.1.3 Post-Loss-of-Coolant Accident Long-Term Subcriticality, Core Cooling Analyses

6.1.3.1 Introduction and Background

The post-LOCA long-term subcriticality and cooling analyses completed as part of the 422V+ fuel transition (Reference 1) were performed consistent with the uprated core power level of 1772 MWt. The following is a summary of the post-LOCA long-term subcriticality and core cooling analyses performed for the fuel transition and stretch uprate.

Westinghouse's position for satisfying the requirements of 10CFR50.46(b)(5), *Long-Term Cooling*, is specified in WCAP-8339 (Reference 14), WCAP-8471-A (Reference 15), and Technical Bulletin NSID-TB-86-08 (Reference 16). The Westinghouse position is that the reactor will remain shut down by boron alone.

6.1.3.2 Description of Analyses and Evaluations

The post-LOCA mixed mean sump boron concentration for the 422V+ fuel transition and stretch uprate and a plot of the containment sump boron concentration as a function of pre-trip Reactor Coolant System (RCS) boron concentration (Figure 6.1-1) were developed. The plot is included in the Reload Safety Analysis Checklist (RSAC) and is checked for each reload cycle to ensure that adequate boron will exist in the sump to maintain subcriticality in the long-term post-LOCA.

In addition to confirming the post-LOCA subcriticality, maintenance of long-term core cooling (LTCC) after switchover to sump recirculation mode must be demonstrated. For the LBLOCA, core cooling is ensured by confirming that there is sufficient ECCS flow to offset core boil-off and boiling in the downcomer and lower plenum. For the SBLOCA, potential effects of ECCS flow interruptions and/or enthalpy changes at switchover to the recirculation mode are considered as part of the SBLOCA analysis. Consistent with the other LOCA analyses, the uprated core power level of 1772 MWt, plus a 0.6-percent allowance for calorimetric uncertainty was analyzed.

6.1.3.3 Acceptance Criteria and Results

For the 422V+ fuel transition and stretch uprate, it was demonstrated that recirculation ECCS flows are sufficient to maintain post-LOCA long-term cooling. The post-LOCA mixed mean sump boron concentration will be checked for each reload cycle to ensure that adequate boron will exist in the sump to maintain subcriticality in the long-term post-LOCA.

6.1.3.4 Conclusions

The LTCC and subcriticality analyses support combinations of core power and uncertainty so that the total core power inclusive of uncertainty does not exceed 1783 MWt.

6.1.4 Post-Loss-of-Coolant Accident Boron Buildup Analysis

6.1.4.1 Introduction and Background

The post-LOCA boron buildup analysis completed as part of the 422V+ fuel transition (Reference 1) was performed consistent with the uprated core power level of 1772 MWt. The following is a summary of the post-LOCA boron buildup analysis performed for the fuel transition and stretch uprate.

6.1.4.2 Description of Analyses and Evaluations

In the long-term post-LOCA, it must be demonstrated that the boron does not precipitate in the reactor core. For plants equipped with UPI like KNPP, precipitation is precluded by ensuring that residual heat removal (RHR) flow into the reactor vessel through the UPI RHRS is established prior to the time boron precipitation is calculated to occur. This is ensured for the LBLOCA scenario since the RCS rapidly depressurizes to a point well below the cut-in pressure of the RHRS. The limiting SBLOCA scenario for KNPP was re-analyzed for the 1772-MWt uprated core power conditions.

Much like the replacement steam generator (RSG) analysis for KNPP, the precipitation point was based on experimentally determined boric acid solubility data at 259°F reduced by 4 weight-percent for uncertainty. This corresponds to a saturation pressure of 35 psia, which is slightly higher than was used in the RSG analysis, but is still conservatively chosen well below the RHR cut-in pressure. Other than the assumed RCS saturation temperature and pressure, all aspects of the calculation were consistent with, or conservative with respect to, the methodology described in CLC-NS-309 (Reference 17). Consistent with the other LOCA analyses, the uprated core power level of 1772 MWt plus, a 0.6-percent allowance for calorimetric uncertainty, was analyzed.

6.1.4.3 Acceptance Criteria and Results

The post-LOCA boron buildup re-analysis for KNPP for the 422V+ fuel transition and stretch uprate demonstrates the continued acceptability of an 18-hour criterion for initiating RHR post-LOCA.

6.1.4.4 Conclusions

The post-LOCA boron buildup analysis supports combinations of core power and uncertainty so that the total core power inclusive of uncertainty does not exceed 1783 MWt.

6.1.5 References

1. R. H. Owoc, *Kewaunee Nuclear Power Plant, RTSR Program*, RTSR, KEW-RTSR-02-021, July 19, 2002, attaches Sisk, R. B., *Reload Transition Safety Report for the Kewaunee Nuclear Power Plant*, July 2002 (NRC Submittal via NMC Letter NRC-02-067, *License Amendment Request 187 to the Kewaunee Nuclear Power Plant Technical Specifications. Conforming Technical Specification Changes for Use of Westinghouse VANTAGE+ Fuel* (Docket 50-305), July 26, 2002.

2. D. J. Wanner (NMC) and J. T. Holly (NMC) to R. H. Owoc (W), *Kewaunee RTSR Input Assumptions and LOCA Input Confirmation – February 2002 Transmittal*, February 15, 2002.

3. WCAP-12945-P-A (Proprietary), Volume I, Rev. 2, and Volumes II-V, Rev. 1, and WCAP-14747 (Non-Proprietary), *Westinghouse Code Qualification Document for Best-Estimate Loss-of-Coolant Accident Analysis*, S. M. Bjorek, et al., March 1998.

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7. WCAP-10079-P-A, *NOTRUMP - A Nodal Transient Small-Break and General Network Code*, P. E. Meyer, August 1985.

8. WCAP-8301, *LOCTA-IV Program: Loss-of-Coolant Transient Analysis*, F. M. Bordelon, et al., June 1974.

9. WCAP-10054-P-A, *Westinghouse Small-Break ECCS Evaluation Model Using the NOTRUMP Code*, N. Lee, et al., August 1985.

10. WCAP-10054-P-A, Addendum 2, *Addendum to the Westinghouse Small-Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model*, Rev. 1, C. M. Thompson, et al., July 1997.

11. WCAP-11145-P-A, *Westinghouse Small-Break LOCA ECCS Evaluation Model Generic Study with the NOTRUMP Code*, S. D. Rupprecht, et al., October 1986.

12. WCAP-14719-P-A, *1-D Heat Conduction Model for Annular Fuel Pellets*, D. J. Shimeck, May 1998.

13. WCAP-10444-P-A, *Reference Core Report Vantage 5 Fuel Assembly*, S. L. Davidson and W. R. Kramer, September 1985.

14. WCAP-8339, *Westinghouse ECCS Evaluation Model Summary*, (Non-Proprietary), F. M. Bordelon, et al., July 1974.

15. WCAP-8471-P-A (Proprietary), and WCAP-8472-A (Non-Proprietary), *Westinghouse ECCS Evaluation Model: Supplementary Information*, F. M. Bordelon, et al., April 1975.

16. Westinghouse Technical Bulletin NSID-TB-86-08, *Post-LOCA Long-Term Cooling: Boron Requirements*, C. M. Thompson, October 31, 1986.

17. CLC-NS-309, *Long-Term Core Cooling – Boron Considerations*, C. L. Caso, April 1, 1975.

Table 6.1-1

KNPP Conditions Analyzed with WCOBRA/TRAC Compared to BE-UPI Test Conditions

Condition	BE-UPI Test	KNPP
Core Power, MWt	1980	1772
Low-Power Region Average Linear Heat Rate (kW/ft)	6.9	1.35 – 4.11
Peak Linear Heat Rate (kW/ft)	17.0	16.995*

*Note that this is a higher kW/ft than the Technical Specifications would allow.

Table 6.1-2

Best-Estimate UPI LBLOCA Results

Result	Value	Criteria
50th Percentile PCT (°F)	<1760	N/A
95th Percentile PCT (°F)	<2084	<2200
Maximum Cladding Oxidation (%)	8.44	<17
Maximum Hydrogen Generation (%)	0.74	<1
Coolable Geometry	Core remains coolable	Core remains coolable
Long-Term Cooling	Core remains cool in long term	Core remains cool in long term

Table 6.1-3

SBLOCA Fuel Cladding Results

Result	Criteria	High T_{avg}[2]			Low T_{avg}
		2-Inch	3-Inch	4-Inch	3-Inch
Peak Clad Temperature (°F)	<2200	916	1030	938	861
Peak Clad Temperature Elevation (ft)	N/A	11.00	11.00	9.75	11.00
Maximum Local Zirc-Water Reaction, (%)	<17.0	<17.0	<17.0	<17.0	<17.0
Maximum Local Zirc-Water Reaction Elevation (ft)	N/A	11.25	11.00	10.25	11.25
Total Zirc-Water Reaction (%)	<1.0	<1.0	<1.0	<1.0	<1.0
Hot Rod Burst Time (sec)	N/A	No burst	No burst	No burst	No burst
Hot Rod Burst Elevation (ft)	N/A	N/A	N/A	N/A	N/A
Reactor Core Rated Thermal Power[1]		1772 MWt			
Peak Linear Power		16.73 kW/ft			
Total Peaking Factor, Fq		2.5			

Notes:

1. 0.6% is added to the core thermal power to account for calorimetric uncertainties.
2. A high T_{avg} 6-inch break case was performed resulting in no core uncovery.

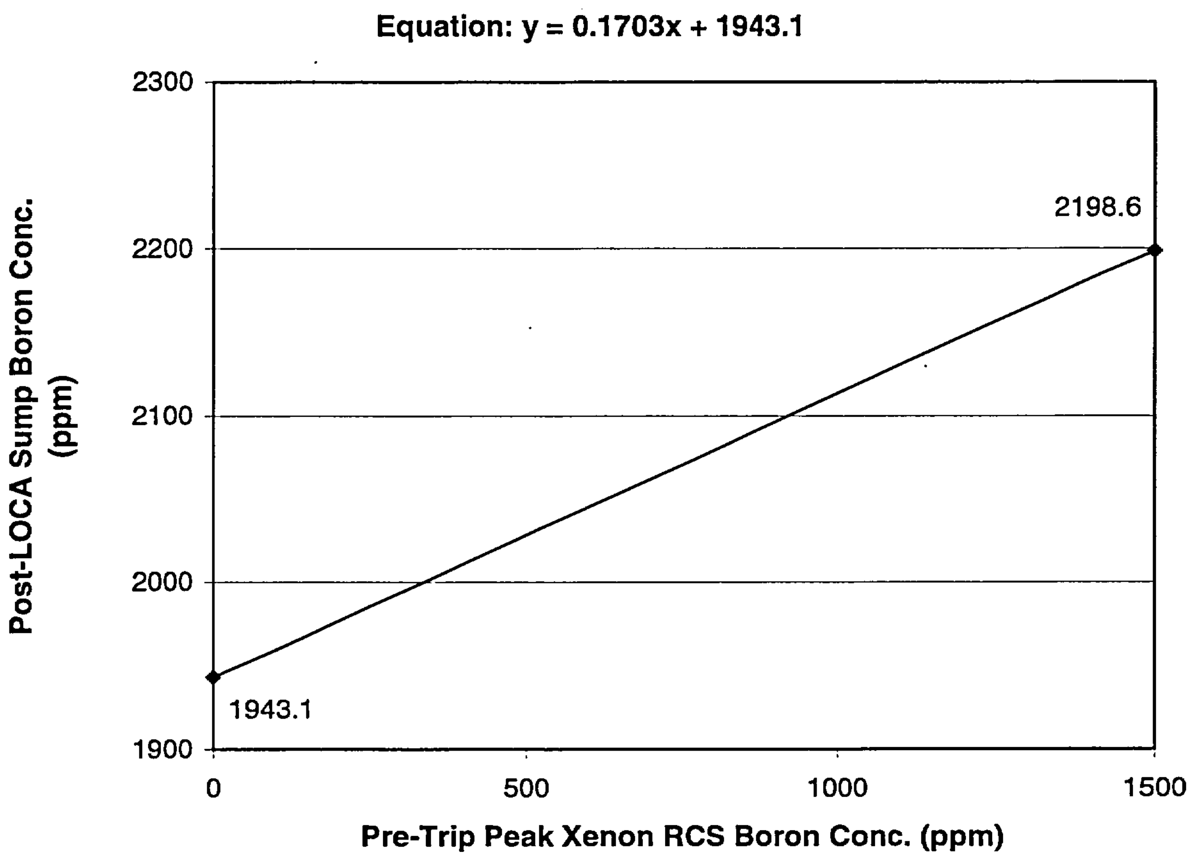

Figure 6.1-1

Kewaunee Post-LOCA Sump Boron Concentration

6.2 Non-Loss-of-Coolant Accident Events

6.2.1 Introduction and Background

The required non-loss-of-coolant accident (non-LOCA) accident analyses of Chapters 14.1 and 14.2 of the *Updated Safety Analysis Report* (USAR) for the Kewaunee Nuclear Power Plant (KNPP) are listed in Table 6.2-1. Note that the transients described in USAR Sections 14.2.1 through 14.2.4 are addressed in Sections 6.3 ("Steam Generator Tube Rupture") and 6.7 ("Radiological Consequences Evaluation") of this report, respectively, and the main steam line break containment response analysis is described in subsection 6.4.2.2. Also, no evaluation is required for USAR Section 14.2.7 because it has previously been deleted.

With the exception of the anticipated transient without scram (ATWS) and loss-of-normal feedwater (LONF) events, all of the non-LOCA transients listed in Table 6.2-1 were previously analyzed or evaluated in support of the KNPP Fuel Upgrade (transition to Westinghouse 422V+ fuel) and 7.4-percent Power Uprate Programs. These analyses and evaluations are documented in the *Reload Transition Safety Report* (RTSR) of Reference 1. Although the RTSR contains an evaluation of the ATWS event with respect to the transition to Westinghouse 422V+ fuel, the 7.4-percent power uprate was not supported. Therefore, the ATWS event had to be analyzed in support of the 7.4-percent power uprate. The ATWS analysis is discussed in subsection 6.2.2, and the LONF analysis is discussed in subsection 6.2.3.

The results of the ATWS and LONF analyses described in subsections 6.2.2 and 6.2.3, along with the results of the analyses and evaluations described in the RTSR (Reference 1) for all other non-LOCA transients, demonstrate that applicable safety analysis acceptance criteria have been satisfied at the uprated conditions detailed in Table 6.2-2.

6.2.2 Anticipated Transients without Scram (Updated Safety Analysis Report Section 14.1.11)

An ATWS is defined as an anticipated operational occurrence (such as a loss of normal feedwater, loss of condenser vacuum, or loss of offsite power) combined with an assumed failure of the Reactor Trip System to shut down the reactor.

The final ATWS rule, 10CFR50.62(c) (Reference 2), requires Westinghouse-designed pressurized water reactors (PWRs), such as the KNPP, to incorporate an actuation device that is diverse from the Reactor Trip System to automatically initiate the Auxiliary Feedwater System

(AFS), as well as initiate a turbine trip for conditions indicative of an ATWS. The installation of an NRC-approved ATWS mitigating system actuation circuitry (AMSAC) satisfies this final ATWS rule. An approved AMSAC system has been installed at Kewaunee as described in Section 14.1.11 of the KNPP USAR and, therefore, the requirements of 10CFR50.62(c) have been satisfied.

In 1998, in response to an engineering evaluation of the AFS, a Diverse Scram System (DSS) was installed at Kewaunee to ensure the auxiliary feedwater (AFW) pumps would continue to run throughout a loss-of-main feedwater ATWS. The DSS, in conjunction with the AMSAC system, will end the transient before the AFW flow to the steam generators increases to a point where AFW pump net positive suction head (NPSH) could be lost. The DSS is initiated on a signal from the existing AMSAC system and de-energizes the rod drive motor/generator (M/G) set exciter field. Removing the rod drive M/G set exciter field will interrupt power to the control rod gripper, allowing the control rods to free-fall into the core, ending the ATWS event.

6.2.2.1 Description of Analysis

The current Kewaunee USAR licensing basis ATWS analysis credits the DSS in conjunction with the AMSAC system for event mitigation. Specifically, the AMSAC system is credited to actuate a turbine trip and the AFW pumps when the AMSAC signal is actuated (satisfies the final ATWS rule), and the DSS is credited to insert the control rods on the AMSAC signal ending the ATWS transient before the AFW flow to the steam generator increases to a point where AFW pump NPSH could be lost.

To address the 7.4-percent power uprate for Kewaunee, the loss-of-normal feedwater ATWS was analyzed to show that the following criteria continue to be met at the Uprated Power conditions:

- The analytical basis for the final ATWS rule (10CFR50.62(c)) continues to be met.

- The steam generator pressure is greater than 640 psig at all times subsequent to auxiliary feedwater pump initiation and prior to the time of reactor trip (see the NRC Safety Evaluation Report of Reference 5). The lowest steam generator pressure, where the available AFW NPSH equals the required NPSH, is 640 psig. Satisfying this acceptance criterion ensures that the AFS can be relied upon to start and operate throughout the ATWS transient.

Each of these criterion was addressed in a separate analysis.

A conservative approach was taken to show that the analytical basis for the final ATWS rule continues to be met, wherein no credit was taken for the DSS. The basis for the final ATWS rule and the AMSAC design are supported by Westinghouse generic analyses reported in NS-TMA-2182 (Reference 3). These analyses were performed based on guidelines published in NUREG-0460 (1978) (Reference 4).

NS-TMA-2182 describes the methods used in the analyses and provides generic analyses for two-loop, three-loop, and four-loop plant designs with several different steam generator models available in plants at the time NS-TMA-2182 (Reference 3) was developed. It was shown that the analysis results for the four-loop plant design (3423 MWt) were more limiting than those for the two- and three-loop plant designs, and it has been demonstrated that the Westinghouse plant designs satisfy the criteria delineated in NUREG-0460 (Reference 4).

The primary input to the analysis addressing the final ATWS rule was the two-loop loss-of-normal feedwater ATWS model from the generic analysis supporting NS-TMA-2182. The nominal and initial conditions were updated to reflect an uprated NSSS power of 1780 MWt, the steam generator data were revised to reflect the Model 54F steam generators, and the AFW flow rate was revised to reflect Kewaunee-specific data. As stated above, no credit was taken for the DSS in the analysis used to demonstrate compliance with the final ATWS rule.

Note that an ATWS occurrence invariably results in an increase in the primary coolant temperature. With a negative moderator temperature coefficient (MTC) in the core, the primary coolant temperature increase will result in an insertion of negative reactivity, which terminates the transient. As such, the MTC is a key input to the ATWS analysis. The MTC value assumed in the generic analyses described in NS-TMA-2182, -8 pcm/°F, was maintained in the ATWS analysis that supports the Kewaunee uprating. As Kewaunee Technical Specification 3.1.f.4 ensures "...the reactor will have a moderator temperature coefficient no less negative than -8 pcm/°F for 95-percent of the cycle time at full power," the basis for the final ATWS rule, as it relates to MTC and unfavorable exposure time (UET), is satisfied. At the time of this writing, note that a Kewaunee license amendment request for a core operating limits report (COLR) is under review by the NRC (Reference 5). With NRC approval of the COLR amendment, KNPP TS 3.1.f.4 will be relocated to the COLR; the requirements for moderator temperature coefficient (MTC) remain the same in the proposed COLR as in the TS 3.1.f.4.

An ATWS analysis was also performed to address the steam generator pressure criterion. Since the DSS was installed at KNPP to address NPSH concerns, the DSS in conjunction with the AMSAC system was credited for event mitigation. The primary sources of input to this analysis were the two-loop loss-of-normal-feedwater ATWS model from the generic analysis supporting NS-TMA-2182 and the NRC Safety Evaluation Report documented in Reference 6. Therefore, in addition to modifying the nominal and initial conditions to reflect an uprated NSSS power of 1780 MWt and revising the steam generator data to reflect the Model 54F steam generators, the DSS assumptions outlined in Reference 6 were incorporated.

6.2.2.2 Results and Conclusions

To remain consistent with the basis of the final ATWS rule (10CFR50.62(c)), the peak Reactor Coolant System (RCS) pressure calculated in the loss-of-normal feedwater ATWS analysis at the uprated NSSS power level of 1780 MWt should be similar to, or less than, the peak RCS pressure calculated in the limiting four-loop loss-of-normal feedwater reference ATWS analyses (Reference 3). The peak RCS pressure documented in NS-TMA-2182 (Reference 3) for the limiting four-loop loss-of-normal feedwater reference ATWS analysis is 2848 psia. This criterion is more restrictive than the ASME Boiler and Pressure Vessel Code Level C service limit stress criterion of 3200 psig. The peak RCS pressure obtained for the loss-of-normal-feedwater ATWS analysis at an uprated NSSS power level of 1780 MWt with Model 54F steam generators is 2747 psia. Thus, it has been demonstrated that the analytical basis for the final ATWS rule continues to be met for operation of Kewaunee with Model 54F steam generators at an uprated NSSS power of 1780 MWt.

The lowest steam generator pressure, where available AFW NPSH equals the required NPSH ensuring that the AFS can be relied upon to start and operate throughout the ATWS transient, is 640 psig (NRC Safety Evaluation Report [Reference 5]). The results of the analysis crediting the DSS in conjunction with the AMSAC system show that the steam generator pressure is greater than 640 psig subsequent to AFW pump initiation and prior to the time of reactor trip. Thus, it has been demonstrated that the Reference 6 steam generator pressure criterion continues to be met for operation of Kewaunee with Model 54F steam generators at an uprated NSSS power of 1780 MWt.

6.2.3 Loss-of-Normal Feedwater (*Updated Safety Analysis Report*, Section 14.1.10)

Refer to Appendix 6A at the end of Section 6 of this document (see page 6-249).

6.2.4 References

1. R. H. Owoc, *Kewaunee Nuclear Power Plant, RTSR Program*, RTSR, KEW-RTSR-02-021, July 19, 2002, attaches Sisk, R. B., *Reload Transition Safety Report for the Kewaunee Nuclear Power Plant*, July 2002 (NRC Submittal via NMC Letter NRC-02-067, *License Amendment Request 187 to the Kewaunee Nuclear Power Plant Technical Specifications. Conforming Technical Specification Changes for Use of Westinghouse VANTAGE+ Fuel* (Docket 50-305), July 26, 2002.

2. ATWS Final Rule – 10CFR50.62 and Supplementary Information Package, *Reduction of Risk from Anticipated Transients without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants.*

3. NS-TMA-2182, *Anticipated Transients without Scram for Westinghouse Plants*, December 1979.

4. NRC Staff Report NUREG-0460, *Anticipated Transients without Scram for Light Water Reactors*, April 1978.

5. NRC-020064 from M. E. Warner to Document Control Desk, *License Amendment Request 185 to the Kewaunee Nuclear Power Plant Technical Specifications, 'Core Operating Limits Report Implementation,' July 26, 2002 (TAC No. MB5717)*

6. NRC Letter from W. O. Long (NRC) to M. L. Marchi (WPSC), *Kewaunee Safety Evaluation – AMSAC Modification (MA0151)*, July 29, 1998.

Table 6.2-1 Non-LOCA Transients for KNPP		
Transient	**USAR Section**	**Report/Section Reference**
Uncontrolled RCCA Withdrawal from a Subcritical Condition	14.1.1	RTSR Section 5.1.1
Uncontrolled RCCA Withdrawal at Power	14.1.2	RTSR Section 5.1.2
RCCA Misalignment (dropped rod)	14.1.3	RTSR Section 5.1.3
CVCS Malfunction	14.1.4	RTSR Section 5.1.4
Startup of an Inactive Reactor Coolant Loop	14.1.5	RTSR Section 5.1.5
Feedwater Temperature Reduction Incident	14.1.6	RTSR Section 5.1.6
Excessive Heat Removal due to Feedwater System Malfunctions (feedwater flow increase)	14.1.6	RTSR Section 5.1.6
Excessive Load Increase Incident	14.1.7	RTSR Section 5.1.7
Loss of Reactor Coolant Flow/Locked Rotor	14.1.8	RTSR Section 5.1.8
Loss of External Electrical Load	14.1.9	RTSR Section 5.1.9
Loss of Normal Feedwater	14.1.10	Uprate Report Section 6.2.3
ATWS	14.1.11	Uprate Report Section 6.2.2
Loss of AC Power to the Plant Auxiliaries	14.1.12	RTSR Section 5.1.11
Steam Line Break	14.2.5	RTSR Section 5.1.12
Rupture of a CRDM Housing (RCCA ejection)	14.2.6	RTSR Section 5.1.13

Table 6.2-2 Non-LOCA Plant Nominal Parameters for the KNPP Power Uprating Program	
Parameter	**Nominal Conditions**
Core Power (MWt)	1772
Nominal Total RCP Heat (MWt) [1]	8.0
Full-Power Vessel T_{avg} (°F) [2] Maximum Minimum	 573.0 556.3
No-Load RCS Temperature (°F)	547.0
RCS Thermal Design Flow (gpm)	178,000
Pressurizer Pressure (psia)	2250

Notes:

1. Total RCP heat input minus RCS thermal losses.
2. A full-power RCS T_{avg} window between 556.3° and 573.0°F is supported.

6.3 Steam Generator Tube Rupture Transient

6.3.1 Introduction and Background

In support of the Kewaunee Power Uprating Program, a steam generator tube rupture (SGTR) thermal-hydraulic analysis for calculation of the radiological consequences has been performed. The analysis was performed using the NSSS design parameters for a 7.4-percent power uprate to a nominal core power of 1772 MWt.

The major hazard associated with an SGTR event is the radiological consequences resulting from the transfer of radioactive reactor coolant to the secondary side of the ruptured steam generator and subsequent release of radioactivity to the atmosphere. The primary thermal-hydraulic parameters which affect the calculation of doses for an SGTR include the amount of reactor coolant transferred to the secondary side of the ruptured steam generator, the amount of primary to secondary break flow that flashes to steam and the amount of steam released from the ruptured steam generator to the atmosphere. The radiological consequences analysis will be discussed in subsection 6.7.5.

6.3.2 Input Parameters and Assumptions

The accident analyzed is the double-ended rupture of a single steam generator tube. It is assumed that the primary-to-secondary break flow following an SGTR results in depressurization of the Reactor Coolant System (RCS), and that reactor trip and safety injection (SI) are automatically initiated on low pressurizer pressure. Loss-of-offsite power (LOOP) is assumed to occur at reactor trip resulting in the release of steam to the atmosphere via the steam generator atmospheric relief valves (ARVs) and/or safety valves. After plant trip and SI actuation it is assumed that the RCS pressure stabilizes and the break flow equilibrates at the point where incoming safety injection flow is balanced by outgoing break flow as shown in Figure 6.3-1. The equilibrium primary-to-secondary break flow is assumed to persist until 30 minutes after the initiation of the SGTR, at which time it is assumed that the operators have completed the actions necessary to terminate the break flow and the steam releases from the ruptured steam generator.

The analysis does not require that the operators demonstrate the ability to terminate break flow within 30 minutes from the start of the event. It is recognized that the operators may not be able to terminate break flow within 30 minutes for all postulated SGTR events. The purpose of the calculation is to provide conservatively high mass-transfer rates for use in the radiological consequences analysis. This is achieved by assuming a constant break flow at the equilibrium flow rate, with a constant flashing fraction that does not credit the plant cooldown, for a relatively long time period. Thirty minutes was selected for this purpose. This modeling is consistent with the SGTR analysis presented in Section 14.2.4 of the current *Updated Safety Analysis Report* (USAR).

After 30 minutes, it is assumed in the analysis that steam is released only from the intact steam generators in order to dissipate the core decay heat and to subsequently cool the plant down to the Residual Heat Removal System (RHRS) operating conditions. It is assumed that plant cooldown to RHR operating conditions is accomplished within 24 hours after initiation of the SGTR and that steam releases are terminated at that time. A primary and secondary side mass and energy balance is used to calculate the steam release and feedwater flow for the intact steam generators from 0 to 2 hours, and from 2 to 24 hours.

The following analysis assumptions and input parameters were used.

- Analysis methodology is consistent with current USAR analysis and is applicable to equilibrium and transition cores.
- LOOP is assumed to occur concurrent with the reactor trip.
- The core power is 1772 MWt.
- The RCS average temperature range is 556.3° to 573.0°F.
- The steam generator tube plugging (SGTP) range is 0 to 10 percent.
- The low-pressurizer pressure SI actuation setpoint is 1829.7 psia.
- The lowest steam generator safety valve reseat pressure is 895.6 psia. This includes an 18-percent main steam safety valve (MSSV) blowdown, which covers the -3-percent safety valve setpoint tolerance.

- The maximum high-head safety injection (HHSI) flow rates are shown below:

RCS Pressure (psia)	HHSI Flow Rate (lbm/sec)
1000.0	136.9
1100.0	131.2
1200.0	125.3
1300.0	119.2
1400.0	112.8
1500.0	106.0
1600.0	99.0
1700.0	91.6
1800.0	83.7
1900.0	72.5
2000.0	60.0
2100.0	45.4
2200.0	24.6
2238.4	0.0

- The RHR cut-in time (termination of steam releases) is 24 hours.

- The break-flow flashing fraction is calculated based on the initial hot leg temperature (606.8°F) with no reduction.

- The duration of the break-flow to ruptured steam generator and steam releases from the ruptured steam generator is 30 minutes.

- The minimum total auxiliary feedwater (AFW) flow rate supplied to the plant is 160 gpm.

6.3.3 Description of Analyses and Evaluations

The SGTR analysis supports an average temperature (T_{avg}) window range of 556.3°F up to 573.0°F. Plant secondary side conditions (for example, steam pressure, flow, and temperature) are based on high and low tube plugging (0-percent up to 10-percent average/peak) to bound all possible conditions. Four separate cases have been analyzed as follows:

1. T_{avg} = 556.3°F and SGTP = 0 percent

2. T_{avg} = 556.3°F and SGTP = 10-percent average/peak

3. T_{avg} = 573.0°F and SGTP = 0 percent

4. T_{avg} = 573.0°F and SGTP = 10-percent average/peak

In total, four cases were considered in the SGTR thermal-hydraulic analysis to bound the operating conditions for the uprate. Note that these four cases are individually analyzed to determine the limiting steam release and limiting break flow between 0 and 30 minutes (break-flow termination) for the radiological consequences calculation.

A portion of the break flow will flash directly to steam upon entering the secondary side of the ruptured steam generator. Since a transient break-flow calculation is not performed for Kewaunee, a detailed time-dependent flashing fraction that incorporates the expected changes in primary side temperatures cannot be calculated. Instead, a conservative calculation of the flashing fraction is performed using the limiting conditions from the break-flow calculation cases. Two time intervals are considered, as in the break-flow calculations: pre- and post-reactor trip (SI initiation occurs concurrently with reactor trip). Since the RCS and steam generator conditions are different before and after the trip, different flashing fractions would be expected.

The flashing fraction is based on the difference between the primary side fluid enthalpy and the saturation enthalpy on the secondary side. Therefore, the highest flashing will be predicted for the case with the highest primary side temperatures. For the flashing-fraction calculations, it is conservatively assumed that all of the break flow is at the hot leg temperature (T_{hot}) (the break is assumed to be on the hot-leg side of the steam generator). Similarly, a lower secondary side pressure maximizes the difference in the primary and secondary enthalpies, resulting in more flashing. The highest possible pre-trip flashing fraction, based on the range of operating conditions covered by this analysis, is for a case with a T_{hot} of 606.8°F, an initial RCS pressure of 2250 psia, and an initial secondary pressure of 634 psia. All cases consider the same post-trip RCS pressure of 1930 psia and post-trip steam generator pressure of 895.6 psia. The highest post-trip flashing fraction, based on the range of operating temperatures covered by this analysis, is for a case with a T_{hot} of 606.8°F. It is conservatively assumed that the T_{hot} is not reduced for the 30 minutes in which break flow is calculated.

A single calculation is performed to determine long-term steam releases from the intact steam generators for the time interval from the start of the event (0 hours) to 2 hours, and from 2 hours to RHR cut-in at 24 hours. The 0- to 2-hour calculations use the 0- to 30-minute intact steam generator steam release and feedwater flow results from the case that resulted in the highest intact steam generator steam and feedwater flow rates.

A simple mass and energy balance is assumed in the calculation of the break flow and steam releases. The energy balance is based on the following assumed conditions at 30 minutes:

- The RCS fluid is at the equilibrium pressure and no-load temperature.
- The pressurizer fluid and steam generator secondary fluid for both the ruptured and intact steam generator is at saturation at no-load temperature.
- The core and clad, primary system metal, pressurizer metal, and steam generator secondary metal are at no-load temperature. Since the RCS fluid is not at a consistent energy state with the ruptured steam generator and the remainder of the primary and secondary systems, energy must be dissipated to reduce the RCS fluid from equilibrium pressure and no-load temperature to saturation at no-load temperature.

It is assumed that the plant is then maintained stable at no-load temperature until 2 hours, and that steam will be released from only the intact steam generator to dissipate the energy from the reduction in the RCS fluid energy state and the core decay heat from 30 minutes to 2 hours.

After 2 hours, it is assumed that plant cooldown to RHR cut-in conditions is initiated by releasing steam from only the intact steam generator. It is assumed that cooldown to RHR cut-in conditions is completed within 24 hours after the SGTR since the cooldown should be accomplished within this time period. After the RHR cut-in conditions are reached, it is assumed that further cooldown is performed using the RHRS, and that the steam release from the intact steam generator is terminated. The energy to be dissipated from 2 to 24 hours is calculated from an energy balance for the primary and secondary systems between no-load conditions at 2 hours, and the RHR entry conditions at 24 hours, plus the core decay heat load from 2 to 24 hours. The amount of steam released from the intact steam generator is calculated from a mass and energy balance for the intact steam generator.

6.3.4 Acceptance Criteria

There are no criteria associated with the thermal-hydraulic calculations. The results of the calculations are used in the determination of the offsite and Control Room dose radiological consequences. Acceptance criteria for offsite and Control Room doses are discussed in subsection 6.7.5.

6.3.5 Results

The tube rupture break flow and ruptured steam generator atmospheric steam releases from 0 to 30 minutes for the four different SGTR cases (discussed in Section 6.3.3) are summarized in Table 6.3-1. Based on the results of these four SGTR cases, bounding values for break flow and steam releases are provided in Table 6.3-2, along with the long-term steam releases, feedwater flows, and steam generator water mass data to be used in radiological consequences analysis. For an SGTR event, the amount of radioactivity released to the atmosphere is highly dependent on the amount of steam released through the safety valves associated with the ruptured steam generator. Therefore, the worst radiological consequences result from the SGTR case with the greatest amount of steam released. Likewise, a greater break flow results in greater radiological contamination of the secondary side that, in turn, results in a greater amount of activity released along with the steam. Maximum break flow and steam release, therefore, represent bounding values that are conservative for an offsite dose evaluation.

The results of the radiological consequences analysis of an SGTR are discussed in subsection 6.7.5.

6.3.6 Conclusions

The SGTR thermal-hydraulic analysis to be used in the radiological consequences calculation has been completed in support of the Kewaunee Power Uprate Program. Subsection 6.7.5 presents the offsite and Control Room dose consequences based in the thermal-hydraulic data in Table 6.3-2.

Table 6.3-1 Case-Specific SGTR Thermal-Hydraulic Results[1]	
Tube Rupture Break Flow for 0 - 30 min.	
T_{avg} = 556.3°F, 0% SGTP	140,629 lbm
T_{avg} = 556.3°F, 10% SGTP	140,736 lbm
T_{avg} = 573.0°F, 0% SGTP	139,842 lbm
T_{avg} = 573.0°F, 10% SGTP	139,977 lbm
Steam Release from Ruptured Steam Generator for 0 - 30 min.	
T_{avg} = 556.3°F, 0% SGTP	69,505 lbm
T_{avg} = 556.3°F, 10% SGTP	68,361 lbm
T_{avg} = 573.0°F, 0% SGTP	78,476 lbm
T_{avg} = 573.0°F, 10% SGTP	77,293 lbm

Note:

1. No margin added.

Table 6.3-2

Bounding SGTR Thermal-Hydraulic Results for Radiological Dose Analysis

Reactor Trip, SI Actuation, and LOOP	173.3 seconds
Pre-Trip (less than 173.3 sec)	
Tube Rupture Break Flow	16,900 lbm
Percentage of Break Flow which Flashes	19.93%
Steam Release Rate to Condenser	1077.8 lbm/sec for each SG
Post-Trip (after 173.3 sec)	
Tube Rupture Break Flow	138,000 lbm
Percentage of Break Flow which Flashes	14.76%
Steam Release from Ruptured Steam Generator up to 2 Hours	86,400 lbm
Steam Released from Intact Steam Generator up to 2 Hours	233,400 lbm
Steam Release from Intact Steam Generator for 2 - 8 Hours	488,800 lbm
Steam Release from Intact Steam Generator for 8 - 24 Hours	662,800 lbm

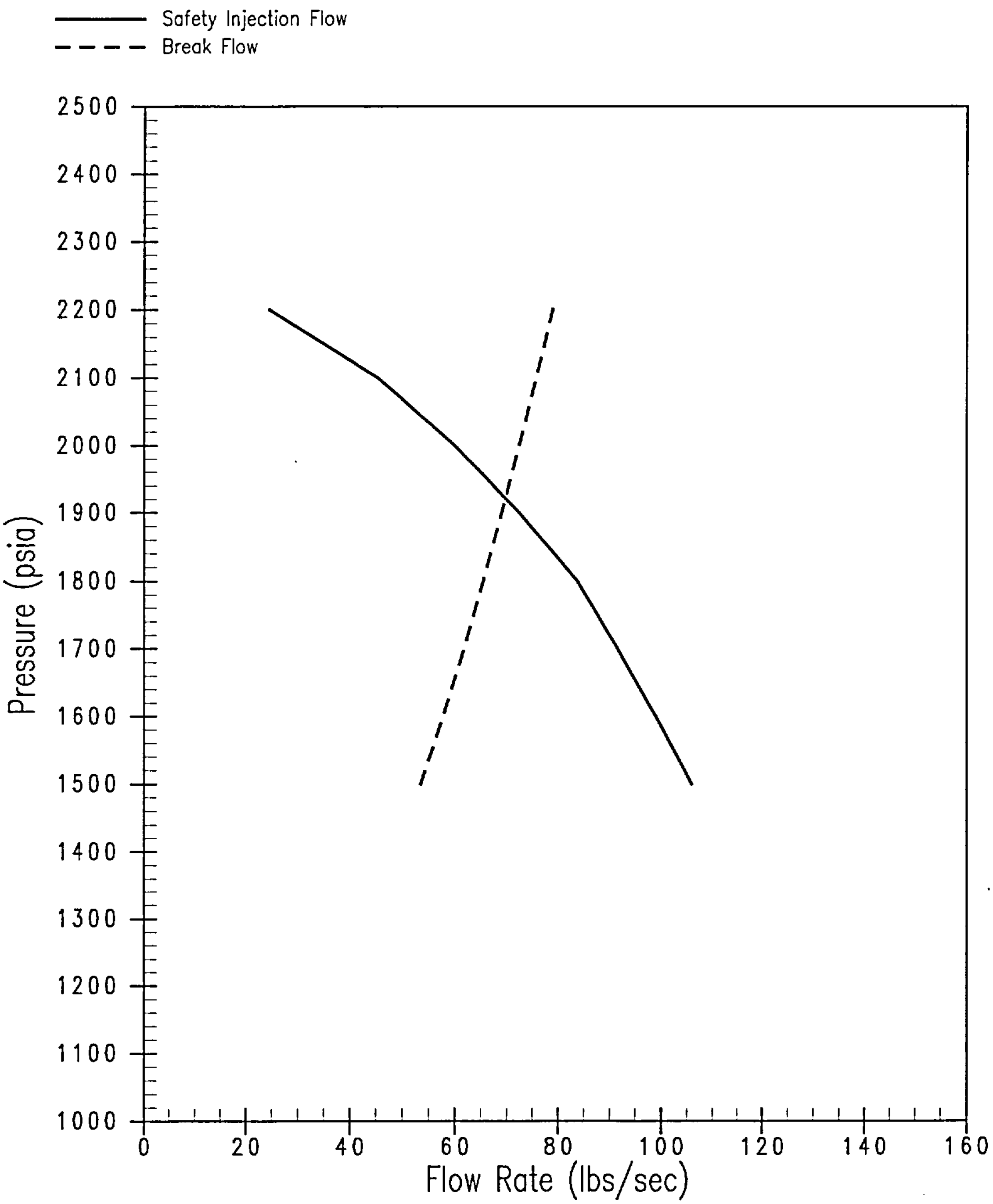

Figure 6.3-1

Safety Injection Flow and Break Flow versus RCS Pressure