Power Rerate Report

for the

Wolf Creek Generating Station

December 1992

Wolf Creek Nuclear Operating Corporation

Executive Summary

The Wolf Creek Nuclear Operating Corporation has prepared this report to obtain NRC approval of the Operating License and Technical Specification changes associated with the Power Rerate Program in conjunction with the allowance for operation over a range of temperatures.

The Wolf Creek Generating Station is currently operating at a Rated Thermal Power (RTP) of 3411 MWth. Upon approval of the associated Technical Specification changes, this RTP will be re-defined to be 3565 MWth. In addition, an analysis has been completed to support operation over a range of temperatures. The current design T_{HOT} is 618.2 ^{O}F. The analysis supports a T_{HOT} reduction of 15 ^{O}F. Thus the allowable operational temperature range extends to a T_{HOT} of 603.2 ^{O}F.

WCGS plans to operate with the power increase and a 5 ^{O}F reduction in T_{HOT}. This report summarizes the evaluations that were performed to confirm the acceptability of these items. The report contains evaluations related to the following:

- Design Operating Conditions
- Nuclear Design and Core Thermal-Hydraulic Design
- LOCA
- Non-LOCA
- Containment Integrity
- RCS Component and Fluid Systems
- Balance of Plant
- Document Revisions
- No Significant Hazards Evaluation
- Environmental Impact Determination

The Wolf Creek Nuclear Operating Corporation has undertaken an extensive analysis effort to support the proposed power uprating of the Wolf Creek Generating Station from 3411 MWth to 3565 MWth rated thermal power. The Power Rerate Report, upon approval of the NRC, will allow the power increase and the operation over a temperature range during Cycle 7 operation, and beyond. This plan was outlined during a meeting with the NRC on April 10, 1991.

The proposed Operating License and Technical Specification changes have been evaluated and determined to be acceptable with respect to all applicable acceptance criteria and that the margin of safety has not been reduced.

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1.0 PROGRAM DESCRIPTION

1.1 DEFINITION OF GOALS

Wolf Creek Nuclear Operating Corporation (WCNOC) has undertaken a program to uprate Wolf Creek Generating Station (WCGS) to a maximum NSSS power level of 3579 MWth. The originally licensed maximum core power level is 3411 MWth, which corresponds to an NSSS power output of 3425 MWth when reactor coolant pump thermal output is included. The uprating program is designed to increase licensed core power to 3565 MWth, with a total NSSS power output of 3579 MWth. Unless otherwise noted, 100% power in this report refers to a core power level of 3565 MWth.

In addition to the uprating, several other plant operating parameter changes are being implemented such as a range in the allowable design reactor vessel outlet temperature (T_{HOT}). This range extends down 15°F from the current T_{HOT} in order to minimize corrosion propensity. The analysis has also incorporated a number of other changes as discussed in the Cycle 7 Technical Specification Change Report (Reference 1). The effects of these changes are discussed in Section 3.0 Safety Evaluations, which support plant operation at any temperature between the nominal and reduced T-HOT condition. Table 1-1 lists the revised plant configuration for the WCGS rerating program.

For the purpose of clarification and distinction, the term "uprating" in this report refers specifically to increased power level, whereas the term "rerating" refers to the power level increase in addition to the sum total of all other parameter changes being incorporated by this program.

The purpose of this report is to provide the basis for the determination that continued safe plant operation can be achieved at the rerated conditions, over a range of temperatures. The licensing basis assessment includes a review of the accident analyses, component design issues related to safety, Emergency Response Guidelines, Technical Specifications and appropriate sections of the USAR. The Power Rerate Report, upon approval of the NRC, will allow the power increase and the operation over a temperature range during Cycle 7 operation, and beyond. This plan was outlined during a meeting with the NRC on April 10, 1991.

References

1. Letter No. NA 92-0073, "Docket No. 50-482: Revision to Technical Specifications for Cycle 7", R. C. Hagan to U. S. NRC.

1.2 APPLICABLE DESIGN CRITERIA

The analyses performed in support of the WCGS rerating program have been completed in accordance with applicable quality assurance requirements. Equipment reviews and evaluations have been performed in accordance with industry codes, standards, and regulatory requirements applicable to WCGS. Assumptions and acceptance criteria for the various analyses are addressed in the respective sections in Section 3.0.

1.3 SCOPE SUMMARY

Section 3.0 of this report provides the evaluations which have been performed to assess the effect of the rerating program on the following systems and components:

- Reactor Vessel
- Reactor Vessel Internals
- Steam Generators
- Control Rod Drive Mechanisms
- Reactor Coolant Pumps
- Pressurizer
- RCS Piping, Supports and Primary Component Supports
- Fluid Systems:
  RCS, chemical and volume control, residual heat removal
- NSSS/BOP Interface:
  main steam system, main feedwater system, auxiliary feedwater, steam generator blowdown, cooling water systems
- Balance of Plant

The following areas have been evaluated to support plant operation at the rerated conditions:

- Design Operating Conditions
- Reactor Core and Thermal-Hydraulic Design
- LOCA
- Non-LOCA
- LOCA and MSLB Mass & Energy Releases
- Containment Integrity
- RCS Component and Fluid Systems
- Balance of Plant
- Document Revisions
- No Significant Hazards Evaluation
- Environmental Impact Determination

Section 2.0 of this report defines the rerated plant design parameters and transients, and summarizes the effect on control systems.

TABLE 1-1

WOLF CREEK GENERATING STATION
RERATING PLANT CONFIGURATION

Item	Plant Target Value
Increased Thermal Power Output (4.5%)	3579 MW_{th}
Reduction in T-HOT	Analysis: 15°F Operation: 5°F
Large Break LOCA Re-Analysis	BASH Model
Small Break LOCA Re-Analysis	NOTRUMP Model
Intermediate Flow Mixers (Start in Cycle 7, Full Core in Cycle 9)	Addition
$F_{\Delta H}$ / F_Q	1.65/2.5
Thimble Plugs	Removal
Positive Moderator Coefficient	+6 pcm/°F
Analysis for Increased Allowable Steam Generator Tube Plugging	10% Avg/10% Peak
OTΔT and OPΔT Trip Setpoints	Optimization
Core Cperating Limit Report	Preparation

2.0 DETERMINATION OF DESIGN OPERATING CONDITIONS

2.1 DESIGN POWER CAPABILITY PARAMETERS

A summary of the design NSSS power capability parameters that have been established is presented in Table 2-1. These parameters have been approved by WCNOC and the Westinghouse Performance Capability Working Group (PCWG). These parameters define the operating conditions for the Rerating Program. The first column lists the current design parameters while the second and third columns list the design parameters at the rerated conditions. Two sets of rerated parameters have been generated to define operation at uprated power and nominal or reduced T-HOT. By assessing the upper and lower limits of the T-HOT range, operation at any T-HOT value in between, when defined for an operating cycle, can be supported. These rerating parameters are those used as initial conditions for the safety, systems and components evaluations and analyses.

It is noted that although the rerating incorporates a 15°F T-HOT reduction, it is not expected that the current turbine design will support the full 15°F reduction at the rated thermal power operation (3579 MWth) due to the volumetric flow limit of the turbine.

2.2 CONTROL AND PROTECTION SYSTEM SETPOINTS

A detailed accounting of the effects of the rerating program on non-safety related control system setpoints has been completed. Changes to these setpoints have been reviewed to assure maintenance of margin between control and protection system setpoints and thus preserve operating flexibility. That is, control systems designed to avoid reactor trips for certain plant transients will continue to perform as intended.

Changes to protection system setpoints as a result of the rerating program have been incorporated into the safety evaluations performed in Section 3.0. These evaluations have concluded that the appropriate safety-related acceptance criteria continue to be met for the rerating program parameters.

2.3 SYSTEM AND COMPONENT DESIGN TRANSIENTS

Revised NSSS design transients have been generated which define the number and degree of pressure and temperature swings for a given set of plant conditions expected to be encountered throughout the plant's design life. These transients form the basis against which the system and component safety evaluations are conducted. As discussed in detail in Section 3.5, all systems and components continue to meet their mechanical design criteria at the rerated conditions.

Not all of the design transients are affected by a change in the full power operating temperatures. In general, only those transients that involve a significant change in reactor power with temperature being controlled by the T_{AVG} program will be affected. Table 2-2 lists the NSSS design transients that have been revised for use in the NSSS systems and components evaluations.

TABLE 2-1

WOLF CREEK GENERATING STATION PERFORMANCE CAPABILITY PARAMETERS

Parameter	Current Design Basis	Rerating (Case 1) Upper Bound T-HOT	Rerating (Case 2) Lower Bound T-HOT
NSSS Power (MW_{th})	3425	3579	3579
Reactor Power (MW_{th})	3411	3565	3565
Thermal Design Flow (gpm/loop)	95700	93600	93600
Reactor Flow, Total (10^6 lbm/hr)	142.1	139.4	142.9
Reactor Coolant Pressure (psia)	2250	2250	2250
Reactor Coolant Temperature (°F)			
Core Outlet	621.4	625.0	608.5
Vessel Outlet	618.2	620.0	603.2
Core Average	591.8	593.0	575.1
Vessel Average	588.5	588.4	570.7
Vessel/Core Inlet	558.8	556.8	538.2
Steam Generator Outlet	558.6	556.6	538.0
Steam Generator			
Steam Temperature (°F)	544.6	538.4	519.4
Steam Pressure (psia)	1000	950	807
Steam Flow, Total (10^6 lbm/hr)	15.14	15.92	15.83
Feedwater Temperature (°F)	440	446	446
Zero Load Temperature (°F)	557	557	557
Steam Generator Tube Plugging (%)	0	10	10
Core Bypass (%)	5.8	8.4	8.4
Fuel Design	17X17	17X17	17X17
	Std.	V-5H*	V-5H*

*With IFMs

TABLE 2-2

WOLF CREEK GENERATING STATION
NSSS SYSTEM DESIGN TRANSIENTS
REVISED FOR RERATING

Transient*

Normal Condition Transients

Unit Loading and Unloading Between 0 and 15% power
Unit Loading and Unloading at 5%/minute
Reduced Temperature Return to Power
Step Load Increase and Decrease of 10%

Upset Condition Transients

Loss of Load
Loss of Power
Partial Loss of Flow
Reactor Trip - Case A
Reactor Trip - Case B
Reactor Trip - Case C
Inadvertent RCS Depressurization
Control Rod Drop
Inadvertent Safety Injection

Emergency Condition Transients

Complete Loss of Flow

* As identified in "System Standard 1.3F NSSS Reactor Coolant System Design Transients Identification No. 1.3F," Westinghouse Nuclear Energy Systems, Water Reactor Division, Revision 0, March 1978.

3.0 SAFETY EVALUATIONS

3.1 NUCLEAR DESIGN AND CORE THERMAL-HYDRAULIC DESIGN EVALUATION

3.1.1 Nuclear Design

The effects of uprating the allowable thermal power and increasing the core peaking factor limits on the nuclear design bases and methodologies for Wolf Creek Generating Station have been evaluated.

The uprated allowable thermal power is 4.5% more than the currently licensed power level. The effects of the allowed thermal power and associated fuel and moderator temperature changes on core physics characteristics are small and are explicitly modeled in the neutronics models. The specific values of core safety parameters, e.g., power distributions, peaking factors, rod worths, reactivity coefficients, are primarily loading pattern dependent. The variations in the loading pattern dependent safety parameters are expected to be typical of the normal cycle to cycle variations for the standard fuel reloads.

In summary, the increase in allowed thermal power from the current level will not reduce the margin of safety in the current WCGS USAR nuclear design bases. However, the design bases will be modified due to the increases to the peaking factor limits and allowed thermal power.

No changes to the nuclear design philosophy or methods are necessary because of the increased allowable thermal power or the use of increased peaking factors. The reload design philosophy includes the evaluation of the reload core key safety parameters which comprise the nuclear design dependent input to the USAR safety evaluation for each reload cycle. These key safety parameters will be evaluated for each reload cycle. If one or more of the parameters fall outside the bounds assumed in the safety analysis, the affected transients will be re-evaluated and the results documented in the reload safety evaluation for that cycle.

3.1.2 Core Thermal-Hydraulic Evaluation

3.1.2.1 Introduction and Summary

This section describes the calculational methods used for the thermal-hydraulic analysis, evaluation of the departure from nucleate boiling (DNB) performance, and the hydraulic compatibility during the transition from mixed fuel cores to an all VANTAGE 5H with intermediate flow mixers (IFM) core. Based on minimal hardware design differences and prototype hydraulic testing of the fuel assemblies, it is concluded that the standard (STD), VANTAGE 5H, and VANTAGE 5H with IFM fuel assembly designs are hydraulically compatible [1]. Table 3.1.2-1 provides a summary of the thermal-hydraulic design parameters for the WCGS that were used in this analysis. The thermal-hydraulic design for the upgraded fuel product was analyzed for an increase in the design limit value for the nuclear enthalpy rise hot channel factor ($F_{\Delta H}$) from 1.55 to 1.65. This increase is achieved by removing unnecessary conservatism in the design through the use of an improved critical heat flux correlation and improved analysis methodologies as described in the following sections. The thermal-hydraulic design criteria and methods remain the same as those presented in the WCGS USAR with the exceptions noted in the following sections. All of the current USAR thermal-hydraulic design criteria are satisfied.

3.1.2.2 Methodology

The existing thermal-hydraulic analysis of the 17x17 STD and VANTAGE 5H fuel used in the WCGS is based on the standard thermal and hydraulic methods and the WRB-1 critical heat flux correlation as described in the Unit USAR and subsequent approved licensing submittals [2]. The DNB analysis of the core containing the 17x17 STD, VANTAGE 5H, and VANTAGE 5H with IFM fuel assemblies has been modified to incorporate the WRB-2 critical heat flux correlation [3] and the Statistical Core Design (SCD) analysis methodology [4,5]. The W-3 correlation and standard methods are still used when conditions are outside the range of the WRB-2 correlation and of the SCD.

3.1.2.2.1 Critical Heat Flux Correlations

The WRB-2 critical heat flux correlation is based entirely on rod bundle data and takes credit for significant improvement in the accuracy of the critical heat flux predictions over previous DNB correlations. As documented in the VANTAGE 5 Fuel Assembly Reference Core Report [3], the WRB-2 correlation was found to predict the CHF test data with a mean measured/predicted ratio of 1.0051 with a standard deviation of 0.0847 over 684 test points. The approved 95/95 limit for DNBR for standard, VANTAGE 5H and VANTAGE 5H with IFM fuel assemblies is 1.17.

The NRC safety evaluation report for VIPRE-01, stated that "use of a CHF correlation which has been previously approved for application in connection with another thermal hydraulic code other than VIPRE-01 will require an analysis showing that, given the correlation data base, VIPRE-01 gives the same or a conservative safety limit." [13]. This section presents a discussion of the qualification effort for the Westinghouse WRB-2 critical heat flux correlation 95/95 design limit in the VIPRE-01 core thermal-hydraulic analysis code.

Qualification of a CHF correlation for use in a thermal-hydraulic analysis code other than the code used to develop the correlation requires that analyses be performed which demonstrates that given the modeling philosophy and correlation set applied, the new code will yield conservative results. This was accomplished for the WRB-2 critical heat flux correlation and the VIPRE-01 code by an analysis of the entire data set used by Westinghouse in the original development of WRB-2 with the THINC code. The VIPRE specific results were then used to calculate the WRB-2 design limit which will insure protection of at least 95 percent of the fuel at a 95 percent confidence level. The calculation of the WRB-2 design limit in VIPRE which provides 95/95 protection is a function of the mean of the measured/predicted critical heat flux ratios, the standard deviation of the measured/predicted ratios, and Owen's one-sided tolerance factor for the data set. The expression for the correlation 95/95 design limit is given by;

$$DNBR_{cl} = \frac{1.0}{M/P - K_{N,\gamma,P} * \sigma_m}$$

Where, $DNBR_{cl}$ = CHF correlation 95/95 design limit.
M/P = Measured to Predicted Ratio
$K_{N,\gamma,P}$ = Owen's one-sided tolerance factor. The factor is a function of the number of members in the correlation data set, the confidence level required, and the percentile protection required.
σ_m = Standard deviation of the measure/predicted ratios

Results from the VIPRE-01, Mod 1 analysis, summarized in Table 3.1.2-2, indicate a mean measured/predicted ratio of 1.022377 and a measure/predicted standard deviation of 0.079859. After adjusting for the degrees of freedom, the standard deviation of the measured/predicted ratios from the VIPRE-01 code increases to 0.080629. The Owen's one-sided tolerance factor for 95% protection at a 95% confidence level is 1.75 for a data set population with 6[illegible] members. Thus, the results of the VIPRE-01, Mod 1 analysis of the WRB-2 correlation data set indicate a 95/95 design limit of;

$$DNBR_{cl} = \frac{1.0}{1.022377 - 1.75(0.080629)}$$

$$DNBR_{cl} = 1.14$$

However, Westinghouse reported the design limit for WRB-2 at 1.17 for use with the THINC code. The wording of the Safety Evaluation Report on the VIPRE-01 code issued by the NRC requires the users of CHF correlations developed for codes other than VIPRE-01 use the same or a more conservative 95/95 design limit. Therefore the correlation design limit to be used for WRB-2 in the VIPRE-01 code is 1.17. Examination of the WRB-2 qualification in VIPRE-01 results reveals two significant points. First, the mean measured/predicted ratio from the VIPRE-01 data of approximately 1.022 indicates that the WRB-2 correlation, as implemented in VIPRE-01, slightly under predicts the experimental critical heat flux as compared to the results originally reported. This is a conservative result. Secondly, the adjusted standard deviation of 0.080629 from the VIPRE-01 qualification results indicates that the under predicting of experimental critical heat flux is consistent across the entire CHF database. Therefore, use of an correlation 95/95 design limit of 1.17 in thermal-hydraulic analyses for the Wolf Creek Generating Station will yield conservative core thermal-hydraulic designs.

The W-3 correlation is used below the first mixing vane grid with a 95/95 limit DNBR of 1.30. The W-3 correlation is also used for the steamline break analyses with a 95/95 limit DNBR of 1.45 in the pressure range of 500 to 1000 psia [6].

3.1.2.2.2 Thermal Design Margins

The determination of the statistical design limit for the rerate of the WCGS, cycle 7, included measurement and calculational uncertainties on five core state variables; power, flow, pressure, temperature, and radial peaking. The core state variables, the associated uncertainties, and the distributions assumed in the determination of the statistical design limit for cycle 7 are summarized in Table 3.1.2-3. The resulting SDL for the WRB-2 critical heat flux correlation for the cycle 7 design was established as 1.31.

Generic design margin, which is used to account for various design penalties, is then added to the statistical design limit. This new DNBR limit, called the thermal design limit (TDL) then defines the acceptance criteria for all DNBR evaluations. For cycle 7, the generic margins allocated were;

Transition Core Penalty	12.0%
Lower Plenum Flow Anomaly	3.0%
Rod Bow Penalty	1.5%
Axial Peaking Uncertainly Penalty	2.5%
Design Margin	8.22%
Total Generic Margin	27.22%

The thermal design limit for the cycle 7 design is then given by the expression;

$$TDL = \frac{SDL}{1.0 - \%Margin/100}$$

$$TDL = \frac{1.31}{[1.0 - (27.22/100)]}$$

$$TDL = 1.80$$

Therefore, analysis results which yield a minimum departure from nucleate boiling ratio greater than or equal to 1.80 will meet the requirement of 95% protection from DNB at the 95% confidence level.

The thermal-hydraulic design criteria is that the probability that DNB will not occur on the most limiting fuel rod is at least 95%, at the 95% confidence level, for any Condition I or Condition II event. Conservative uncertainty values on the treated core state variables are used in SCD to establish overall uncertainty on DNB [5]. SCD analyses use a new flow parameter, termed minimum measured flow (MMF), which is equal to the thermal design flow (TDF) plus the flow uncertainty. Analyses by standard methods continue to use the thermal design flow.

3.1.2.3 Hydraulic Compatibility

Fuel assembly lift forces are defined as the net upward force acting on the assembly due to interaction with coolant flow, excluding fuel assembly weight and buoyancy. Fuel assembly lift forces are used in the design of fuel assembly hold-down springs and reactor vessel internals.

Lift forces are calculated at hot full power, cold startup and hot pump overspeed. Designing to these conditions ensures that the hold-down spring design criterion is met.

The Wolf Creek Generating Station will transition to VANTAGE 5H fuel assemblies with IFM grids from a core consisting of VANTAGE 5H and Standard fuel assemblies. Consequently, lift forces for all three fuel types were evaluated. Based on this evaluation it was concluded that the hydraulic load on the hold-down spring and the core internals for Standard, VANTAGE 5H and the VANTAGE 5 fuel assemblies with IFM grids is acceptable.

3.1.2.4 Core Bypass Flow

Core bypass flow is defined as the total amount of reactor coolant flow which bypasses the core region. This flow is not considered effective in the core heat transfer process. The principal bypass flow paths are:

a. Baffle/Barrel Region
The baffle/barrel region consists of vertical baffle plates that follow the periphery of the core. These are joined to the core barrel by horizontal former plates spaced along the elevation of the baffle plates. The fraction of total flow that passes upward through this region between the core barrel and baffle plates is considered core bypass flow. This bypass flow fraction is large enough to maintain closure head fluid temperature equal to the reactor vessel inlet temperature.

b. Vessel Head Cooling Spray Nozzles

These nozzles are flow paths between the reactor vessel and core barrel annulus and the fluid volume in the vessel closure head region above the upper support plate. A fraction of the flow that enters the vessel inlet nozzles and into the vessel/barrel downcomer passes through these nozzles and into the vessel closure head region.

c. Core Barrel - Reactor Vessel Outlet Nozzle Gap

Some of the flow that enters the vessel/barrel downcomer will leak through the gaps between the core barrel outlet nozzles and the reactor vessel outlet nozzles and merge with the vessel outlet nozzle flow.

d. Fue' Assemoly - Baffle Plate Cavity Gap

This is the core bypass flow path between the peripheral fuel assemblies and the core baffle plates.

e. Fuel Assembly Thimble Tubes

These tubes are physically part of each fuel assembly skeleton and flow within them is partially effective in removing core heat. However, such flow is analytically not considered to be effective in heat removal, and is consequently treated as core bypass flow. Thimble plugging devices reduce this component of the overall bypass flow.

The design value of overall core bypass flow used in existing safety analyses for the Wolf Creek Generating Station is 5.8%. The change in core bypass flow resulting from the VANTAGE 5H fuel assembly with IFM grids was calculated. The result is an increase in the core bypass flow to 6.4% for thimble plugs in place or 8.4% with thimble plugs removed. The best estimate bypass flow, which neglects calculational uncertainties for the case with thimble plugs removed was determined to be 6.61%.

3.1.2.5 Effects of Fuel Rod Bow on DNBR

The phenomenon of fuel rod bowing must be accounted for in the DNBR safety analysis of Condition I and Condition II events. In the IFM region of a VANTAGE 5H with IFM fuel assembly, the grid-to-grid spacing is approximately 10 inches compared to approximately 20 inches in the current fuel assemblies in the Cycle 6 core. Using approved methodology [7], the predicted channel closure in the 10 inch spans in the VANTAGE 5H with IFM assemblies will be less than 50%. Thus, no rod bow penalty is required in this region. In the spans below the IFM region of the VANTAGE 5H with

IFM assemblies and for the resident fuel, rod bow is accounted for in available DNBR margin as summarized in section 3.1.2.2.2.

The maximum rod bow penalties accounted for in the design safety analyses are based on an assembly average burnup of 24,000 MWD/MTU, as approved by the Commission [8]. At burnups greater than 24,000 MWD/MTU, credit is taken for the effect of $F_{\Delta H}$ burndown, due to the decrease in fissionable isotopes and buildup of fission product inventory. No additional rod bow penalty is required.

3.1.2.6 Fuel Temperature Analysis

The 0.374 inch O.D. fuel rod used in the VANTAGE 5H fuel assembly with IFM grids is the same as that used in the Standard and VANTAGE 5H fuel assemblies resident in the core. Fuel performance evaluations are completed for each fuel region to demonstrate that the design criteria will be satisfied for all fuel regions under the planned operating conditions for each reload core. Fuel rod design evaluations are performed using approved models [9, 10, 11]. There is no change in the fuel temperature design criteria used in the safety analysis calculations between the fuel types resident for cycle 6 and the VANTAGE 5H with IFM grids to be loaded for cycle 7.

3.1.2.7 Transition Core Effects

The fuel to be loaded for cycle 7 has IFM grids located in spans between mixing vane grids in the upper region of the fuel assembly. The resident fuel, both the Standard and VANTAGE 5H assemblies, does not feature these intermediate grids. The additional grids introduce localized flow redistribution from the VANTAGE 5H with IFM assembly into the Standard and VANTAGE 5H assemblies at axial zones near the IFM grid positions in a transition core. Between the IFM grids, flow returns to the VANTAGE 5H with IFM assemblies due to the tendency for velocity equalization in parallel open channels. This localized flow redistribution actually benefits the Standard and VANTAGE 5H assemblies. This benefit more than offsets the slight mass flow bias due to velocity equalization at non-gridded locations. Thus, the analysis for a full core of these fuel assembly types remains appropriate for that fuel in a transition core.

Transition cores are analyzed as if they were a full core of one assembly type, VANTAGE 5H with IFM grids in this application. A transition core penalty is then applied to the thermal-hydraulic design analyses to account the impact of the flow redistribution. For VANTAGE 5H with IFM grids, the transition core penalty is a function of the number of VANTAGE 5H with IFM grid assemblies present in the core and is determined using approved methodologies [12]. The transition core penalty for Cycle 7 operation has been established at 12.0%. This penalty is included in the safety analysis limit DNBR such that sufficient margin over the design limit DNBR exists to accommodate the transition core penalty along with other appropriate DNBR penalties.

3.1.2.8 Conclusion

The thermal-hydraulic evaluation of the fuel upgrade and peaking factor increase for the Wolf Creek Generating Station has shown that 17x17 Standard, VANTAGE 5H, and VANTAGE 5H with IFM grids are hydraulically compatible and that the DNB margin gained through the use of the Statistical Core Design methodology and the WRB-2 critical heat flux correlation is sufficient to allow an increase in the design $F_{\Delta H}$ from 1.55 to 1.65. More than sufficient DNBR margin exists in the safety limit DNBR to cover the rod bow and transition core penalties. All thermal-hydraulic design criteria are satisfied.

Table 3.1.2-1

Thermal and Hydraulic Design Parameters - Wolf Creek Generating Station

Thermal and Hydraulic Design Parameters	Parameter Values
Reactor Core Heat Output (MW_{th})	3565
Reactor Core Heat Output (10^6 BTU/HR)	12,165
Heat Generated in Fuel (%)	97.4
Pressurizer Pressure, Nominal (psia)	2250.0
Design Radial Power Distribution*	1.65[1.0 + 0.3(1.0 - P)]
DNB Correlations**	WRB-2 W-3
HFP Nominal Coolant Conditions	
Vessel Thermal Design Flow Rate, including bypass, (GPM)	374,400
Vessel Minimum Measured Flow Rate, including bypass, based on 2.5% flow uncertainty (GPM)	384,000
Core Flow Rate, excluding bypass, based on Thermal Design Flow (GPM)***	342,950
Core Flow Rate, excluding bypass, based on MMF and best estimate bypass flow (GPM)***	358,618
Core Flow Area (ft^2)	51.28
Core Inlet Mass Flux, based on TDF (10^6 lbm/hr-ft^2)	2.517
Core Inlet Mass Flux, based on MMF (10^6 lbm/hr-ft^2)	2.632
Nominal Vessel/Core Inlet Temperature (°F)	549.3
Vessel Average Temperature (°F)	581.2
Core Average Temperature (°F)	585.7
Vessel Outlet Temperature (°F)	613.2
Core Outlet Temperature (°F)	618.4
Average Temperature Rise in Vessel (°F)	63.9
Average Temperature Rise in Core (°F)	69.1

* 1.65 represent 1.59 plus 4% measurement uncertainty.

** W-3 is used for conditions outside the range of applicability of the WRB-2 correlation or the Statistical Core Design.

*** Design bypass flow, including uncertainty is 8.4% with thimble plugs removed. Best estimate bypass flow with thimble plugs removed is 6.61%.

Table 3.1.2-1 - Continued

Thermal and Hydraulic Design Parameters

Heat Transfer	
Active Heat Transfer Surface (ft^2)	59,742
Average Linear Power (kW/ft)	5.69
Peak Linear Power for Normal Operation (kW/ft)*	14.2
Temperature at Peak Linear Power for Prevention of Centerline Fuel Melt (°F)	4700

* Based on maximum F_Q of 2.5.

Table 3.1.2-2

WRB-2 Qualification Results

Category	# of Points	Sample Mean M/P	Standard Deviation
Total CHF Points	684	1.022377	0.079859
Axial Heat Flux			
Uniform	357	1.0327	0.0821
Non-Uniform	327	1.0111	0.0757
Bundle Arrays			
5x5, 0.374" O.D.	501	1.0247	0.0804
5x5, 0.360" O. D.	183	1.0161	0.0781
Subchannel Types			
Typical Cell	508	1.0243	0.0786
Thimble Cell	176	1.0167	0.0831
Heated Lengths			
96" (8 ft.)	213	1.0266	0.0835
168" (14 ft.)	471	1.0205	0.0781
Mixing Vane Grid Spacing			
10"	82	1.0038	0.0725
20"	101	1.0261	0.0810
22"	282	1.0182	0.0701
26"	219	1.0329	0.0912

Table 3.1.2-3

Summary of Uncertainties included in SDL

Variable	Description	Uncertainty	Dist.
Q	Heat Balance	2.0%	Normal
W	RCS Flow	2.5%	Uniform
W	Bypass Flow	1.79%	Uniform
P	Pressurizer Pressure	30.0 psia	Uniform
T	Temperature Control	4.85 °F	Uniform
R	Radial Peaking - Measurement	5.0%	Normal
R	Hot Channel Factors	3.0%	Normal
R	Initial Bundle Spacing	1.5%	Uniform
D	WRB-2 Correlation	0.1479 DNBR	Normal
D	VIPRE-01 Code	5.0%	Normal
D	RSM to VIPRE-01 Fit	4.5%	Normal

3.1.2.9 References

1. "Vantage 5H Fuel Assembly", WCAP-10444-P-A, Addendum 2, Davidson, S. L. ed. et al., April 1988 and Letter from W. J. Johnson (Westinghouse) to M. W. Hodges (NRC), "Supplemental Information for WCAP-10444-P-A Addendum 2, 'VANTAGE 5H Fuel Assembly'", NS-NRC-88-3363, dated July 29, 1988.

2. Letter from Reckley, W. D. (NRC) to Withers, B. D. (WCNOC), "Wolf Creek Generating Station - Amendment No. 51 to Facility Operating License No., NPF-42 (TAC No. 79923), dated November 6, 1991.

3. "Vantage 5 Fuel Assembly Reference Core Report", WCAP-10444-P-A, Davidson, S. L. ed. et al., September 1985.

4. "Statistical Core Design for Mixing Vane Cores", BAW-10170-P-A, Farnsworth, D. A. and Meyer, G. A., Babcock & Wilcox Co., August 1987.

5. "Core Thermal-Hydraulic Analysis Methodology for the Wolf Creek Generating Station", TR-90-0025 W01, Kennamore, W. S. et al., Wolf Creek Nuclear Operating Corporation, July 1990.

6. Letter from Thadani, A. C. (NRC) to Johnson, W. J. (Westinghouse), "Acceptance for Referencing of Licensing Topical Report, WCAP-9226-P/WCAP-9227-NP, Reactor Core Response to Excessive Secondary Steam Releases", dated January 31, 1989.

7. "Fuel Rod Bow Evaluation", Skaritka, J. (Ed.), WCAP-8691, Revision 1, July, 1979.

8. Letter from Berlinger, C. (NRC) to Rahe, E. P. Jr. (Westinghouse), "Request for Reduction in Fuel Assembly Burnup Limit for Calculation of Maximum Rod Box Penalty", dated June 18, 1986.

9. "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations", Weiner, R. A. et al., WCAP-10851-P-A, August 1988.

10. "Improved Analytic Model Used in Westinghouse Fuel Rod Design Computations", Miller, J. V. (Ed.), WCAP-8785, October 1975.

11. "Extended Burnup Evaluation of Westinghouse Fuel", Davidson, S. L. (Ed.) et al., WCAP-10125-P-A, December 1985.

12. "Extension of Methodology for Calculating Transition Core DNBR Penalties", Schueren, P. and McAtee K. R., WCAP-11837-P-A, January, 1990.

13. Letter from Rossi, C. E. (NRC) to Blaisdell, J. A. (UGRA), "Acceptance fro Referencing of Licensing Topical Report, EPRI NP-2511-CCM, 'VIPRE-01: A Thermal-Hydraulic Analysis Code for Reactor Cores', Volumes 1, 2, 3, 4", dated May 1, 1986.

3.2 LOCA AND LOCA-RELATED EVALUATIONS

A loss-of-coolant accident (LOCA) is the result of a pipe rupture of the reactor coolant system (RCS) pressure boundary. For the analyses reported here, a small break is defined as a rupture of the RCS piping with a cross-sectional area less than 1.0 ft^2, in which the normally operating charging system flow is not sufficient to sustain pressurizer level and pressure. This event is considered an American Nuclear Society (ANS) Condition III event which are faults which may occur very infrequently during the life of a plant. A major break (large break) is defined as a rupture with a total cross-sectional area equal to or greater than 1.0 ft^2. This event is considered an ANS Condition IV event, a limiting fault, in that it is not expected to occur during the life of the Wolf Creek Generating Station, but is postulated as a conservative design basis.

The Acceptance Criteria for the LOCA are described in 10 CFR 50.46 (Reference 1) as follows:

A. The calculated peak fuel element clad temperature shall not exceed 2200°F.

B. The amount of fuel element cladding that reacts chemically with water or steam to generate hydrogen, shall not exceed 1% of the total amount of Zircaloy in the fuel rod cladding.

C. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. The localized cladding oxidation limit of 17% is not exceeded during or after quenching.

D. The core remains amenable to cooling during and after the break.

E. The core temperature is reduced and decay heat is removed for an extended period of time, as required by the long-lived radioactivity remaining in the core.

The criteria were established to provide a significant margin in emergency core cooling system (ECCS) performance following a LOCA. WASH-1400 (USNRC 1975) (Reference 2) presents a study in regards to the probability of occurrence of RCS pipe ruptures.

References

1. "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors," 10 CFR 50.46 and Appendix K of 10 CFR 50, Federal Register 1974, Volume 39, Number 3.

2. U.S. Nuclear Regulatory Commission 1975, "Reactor Safety Study - An Assessment of Accident Risks in U.S. commercial Nuclear Power Plants," WASH-1400, NUREG-75/014.

3.2.1 Small Break LOCA

Please refer to Letter No. NA 92-0073, "Docket No. 50-482: Revision to Technical Specifications for Cycle 7", R. C. Hagan, to U. S. NRC.

3.2.2 Large Break LOCA

Please refer to Letter No. NA 92-0073, "Docket No. 50-482: Revision to Technical Specifications for Cycle 7", R. C. Hagan, to U. S. NRC.

3.2.3 <u>LOCA Hydraulic Forcing Functions</u>

This section addresses the reactor vessel LOCA hydraulic forcing function analysis performed in support of the rerating program at WCGS.

3.2.3.1 Introduction

The purpose of the LOCA hydraulic forces analysis is to generate the LOCA hydraulic forcing functions for the rerating. The LOCA hydraulic forcing functions are used in the verification of the structural integrity of the vessel internals and core components. The selection of breaks postulated in the analysis was based on the licensing of the primary reactor coolant system piping with leak-before-break technology. Leak-before-break (LBB) technology exempts the primary reactor coolant system piping rupture from consideration as the design basis for defining structural loads. Consequently, the 144 in^2 reactor vessel inlet nozzle and 144 in^2 reactor vessel outlet nozzle breaks previously postulated in the dynamic analysis (see Reference 2) need no longer be considered. The LOCA hydraulic forces analysis takes advantage of the elimination of the large, primary reactor coolant system pipe ruptures to reduce the expected increase in magnitude of the peak forces which may occur due to the uprated conditions and reduction in reactor coolant temperature. As such, the LOCA hydraulic forcing functions were generated for a postulated rupture of the accumulator branch line in the cold leg and the pressurizer surge line in the hot leg. These are the two largest branch lines connected to the reactor coolant system.

3.2.3.2 Method of Analysis

The MULTIFLEX 1.0 computer program (Reference 1) was utilized to generate the transient hydraulic forcing functions on the vessel and internals due to a postulated pipe rupture in the reactor coolant system. MULTIFLEX 1.0 is an engineering design tool that is used to analyze the coupled fluid-structural interactions in a PWR system during the transient following a postulated pipe rupture in the reactor coolant system. The thermal-hydraulic portion of the MULTIFLEX code is based on the one-dimensional homogeneous model expressed in a set of mass, momentum, and energy conservation equations. These equations are quasi-linear first order partial differential equations solved by the method of characteristics. The employed numerical method utilizes an explicit time scheme along the respective characteristics. MULTIFLEX considers the interaction of the fluid and structure simultaneously, whereby the mechanical equations of vibration are solved through the use of the modal analysis technique. MULTIFLEX 1.0 generates the input for the LATFORC and FORCE2 computer codes.

The LATFORC computer code utilizes MULTIFLEX generated field pressures, together with geometric vessel information (component radial and axial lengths), to determine the horizontal forces on the vessel wall, core barrel, and thermal shield (if applicable).

The LATFORC code represents the downcomer region with a model that is consistent with the model used in the MULTIFLEX blowdown calculation. The downcomer annulus is subdivided into cylindrical segments, formed by dividing this region into circumferential and axial zones. The results of the MULTIFLEX/LATFORC analysis of the horizontal forces are typically stored on magnetic tape and are calculated for the initial 500 msec of the blowdown transient. These forcing functions serve as required input for the structural groups who utilize it in determining the resultant mechanical loads on primary equipment and loop supports, vessel internals, and fuel grids.

The FORCE2 program calculates the hydraulic forces which the fluid exerts on the vessel internals in the vertical direction by utilizing a detailed geometric description of the vessel components and the transient pressures, mass velocities, and densities computed by the MULTIFLEX code. The analytical basis for the derivation of the mathematical equations employed in the FORCE2 code is the conservation of linear momentum (one-dimensional). Note that the computed vertical forces do not include body forces on the vessel internals, such as deadweight or buoyancy. When the vertical forces on the reactor pressure vessel internals are calculated, pressure differential forces, flow stagnation on and unrecoverable orifice losses across, and friction losses on the individual components are considered. These force types are then summed together, depending upon the significance of each, to yield the total vertical force acting on a given component.

3.2.3.3 Results

The new LOCA hydraulic forcing functions generated for the accumulator and pressurizer surge line breaks for the rerating were used as input to analyses of the structural integrity of the reactor vessel internals and core components. The evaluation of structural integrity for the vessel internals and core components can be found in Section 3.5.1.3 of this report. The LOCA hydraulic forcing functions were significantly lower in magnitude than those generated previously as input to the dynamic analysis of Reference 2.

3.2.3.4 References

1. WCAP-8708 "MULTIFLEX, a FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics," 1977

2. WCAP-9643 "Dynamic Analysis of Reactor Pressure Vessel for Postulated Loss-of-Coolant Accidents: SNUPPS Projects," 1979

3.2.4 Hot Leg Switchover to Prevent Boron Precipitation

A calculation of the time to realign the ECCS for hot leg injection to prevent boron precipitation in the event of a LOCA is performed for inclusion in the Plant Emergency Operating Procedures. This time is dependent on power level and the RCS, RWST, and accumulator water masses and boron concentrations. A power level of 1.02 times 3565 MWth was assumed. Further, the RWST Tech Spec maximum boron concentration of 2700 ppm was assumed for all boron sources. A switchover time of 10 hours was calculated.

At the time of switchover, sufficient ECCS flow must be available to prevent additional core uncovery. The Westinghouse commitment is that delivered flow will match 1.5 times the calculated boil-off rate at switchover in order to account for possible entrainment of ECCS flow which does not reach the vessel. This minimum flow has been determined as 39 lb/sec. The ability of the ECCS to deliver this flow, in hot leg injection mode has been verified by WCNOC.

3.2.5 Post-LOCA Long Term Cooling - Subcriticality Requirement

The Westinghouse Evaluation Model commitment is that the reactor will remain shutdown by borated ECCS water residing in the containment sump after a postulated design basis LOCA. No credit is taken for the control rods since the forces associated with the large break LOCA may prevent the rods from fully inserting. As such, borated ECCS water provided by the accumulators and the RWST must have a concentration that, when mixed with other water sources, will result in the core remaining subcritical assuming all control rods out. For post-LOCA long-term core cooling, sump boron concentration is determined by the accumulation of all potential water sources in the containment based on each respective source boron concentration.

The post-LOCA subcriticality requirement is verified on a cycle-specific basis in accordance with References 1 and 2 as part of the reload process.

3.2.6 References

1. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

2. "Design Interface Procedure Between Wolf Creek Nuclear Operating Corporation and Westinghouse Electric Corporation for Wolf Creek Generating Station," September 16, 1991.

3.3 NON-LOCA

Please refer to Letter No. NA 92-0073, "Docket No. 50-482: Revision to Technical Specifications for Cycle 7", R. C. Hagan, to U. S. NRC.

3.3.1 Reduction in Hot Leg Temperature

It has been proposed that the plant may operate with a 5 °F to 15 °F reduction in hot leg temperature from the current design basis in order to extend the steam generator life. This reduction in T_{hot} presents the possibility of pressurizer overfilling in transients characterized by long term heat-ups. Pressurizer overfilling and the relief of water through the safety valves constitutes a violation to Condition II acceptance criteria by generating a more serious plant condition without other faults occurring independently.

Analyses were performed in conjunction with the power rerate effort to quantify the effect of a 15 °F reduction in rerate hot leg temperature from the current design basis for the following long term heat-up events:

- Loss of Non-Emergency AC Power, USAR Section 15.2.6
- Loss of Normal Feedwater, USAR Section 15.2.7
- Uncontrolled RCCA Bank Withdrawal at Power, USAR Section 15.4.2

The analyses showed that no pressurizer overfilling occurred throughout the transients and therefore they are acceptable for the previously discussed T_{HOT} reduction.

3.4 CONTAINMENT INTEGRITY ANALYSES

Containment integrity analyses are performed during nuclear plant design to ensure that the pressure inside containment will remain below the containment building design pressure if a LOCA or a Main Steam Line Break (MSLB) inside containment should occur during plant operation. The analysis ensures that the containment heat removal capability is sufficient to remove the maximum possible discharge of mass and energy to containment from the Nuclear Steam Supply System without exceeding design pressure. The analysis can also be used as a basis for the containment leak rate test pressure to ensure that dose limits will be met in the event of release of radioactive material to containment.

The purpose of this analysis is to provide the analytical basis for operation of WCGS. Specifically, the containment integrity analysis will demonstrate the acceptability of operation by demonstrating that the containment peak pressure resulting from a design basis large break LOCA event and MSLB event will not exceed design pressure.

The analyses presented in this section address the consequences of the mass and energy that is released to containment as a result of a design basis LOCA and a design basis MSLB. The mass and energy release data is subsequently used to verify, via calculations, that the containment design pressure is not exceeded in the event of LOCA or a MSLB. In this manner, the analysis results demonstrate that the containment integrity has not been compromised. Sections 3.4.1 and 3.4.2 present the short and long term mass and energy release evaluations, and Section 3.4.3 presents the results of the containment integrity response calculations following a postulated LOCA and MSLB.

Bounding initial temperatures and pressures for the containment integrity analyses were selected to envelope the limiting conditions for operation. In this manner, the most limiting conditions for operation at full power (3579 MWth) were conservatively chosen.

3.4.1 LOCA Mass and Energy Releases - Short Term

3.4.1.1 Background

Containment subcompartment analyses are performed to demonstrate the adequacy of containment internal structures and attachments when subjected to dynamic localized pressurization effects that occur during the first 3 seconds following a design basis pipe break accident. Subsequent to the postulated rupture, the pressure builds up at a faster rate than the overall containment pressure, thus imposing differential pressure across the walls of the structure.

As part of this evaluation, short term LOCA mass and energy releases, which support the Bechtel subcompartment analyses are addressed.

3.4.1.2 Short Term Mass and Energy Release Evaluation

The current short term LOCA mass and energy releases, Reference 1, were generated with the Westinghouse 1975 M&E release model, Reference 2. Currently, the RCS loop breaks include the 763 in^2 double ended cold leg break and the 436 in^2 double ended pump suction break. There is a penalty of increased releases associated with the rerating if only the RCS initial temperature condition changes are considered. However, if leak-before-break (LBB) is credited, Reference 3, the RCS loop breaks are eliminated from consideration, and the smaller RCS nozzle breaks become limiting. The break sizes associated with the surge line, RHR, and the accumulator nozzles, i.e., < 197 in^2, are significantly less than the RCS loop breaks. The mass and energy releases from the smaller RCS nozzle breaks more than offset the initial RCS condition penalties associated with the rerating. Therefore, the current M&E releases for the RCS loop breaks remain bounding for the rerated conditions with LBB for the loop credited.

The mass and energy releases from the surge line break are currently bounding for the pressurizer compartment. The mass and energy releases from the surge line break are strongly affected by the initial temperature conditions of the fluid. The short term mass and energy releases are linked directly to the critical mass flux, which increases with decreasing temperatures. Since the initial temperatures have decreased for the rerating, the peak critical mass flux and energy releases will increase by 15% as a result of the rerating.

3.4.1.3 Summary

The current short term LOCA mass and energy releases for the RCS loop breaks remain bounding for the rerated conditions with LBB for the loop credited. For the surge line break, the peak mass and energy releases will increase by 15% as a result of the rerating. The short term LOCA mass and energy releases in Reference 1 should be multiplied by a factor of 1.15.

3.4.1.4 References

1. Wolf Creek Updated Safety Analysis Report.

2. WCAP-8312-A, Rev. 2, "Westinghouse Mass and Energy Release Data for Containment Design," August 1975.

3. NUREG-0881-Suppl. 5.

3.4.2 LOCA Mass and Energy Release Analysis - Long Term

3.4.2.1 Purpose

This report section presents the long term LOCA mass and energy releases that were generated in support of the Uprating/T-Hot Reduction Rerating program effort for Wolf Creek.

3.4.2.2 Assumptions/Initial Conditions

The evaluation model for the long term LOCA mass and energy release calculations used was the March 1979 model described in Reference 1. This evaluation model has been reviewed and approved by the NRC, and has been used in the analysis of other dry containment plants.

For the long term mass and energy release calculations, operating temperatures for the highest average coolant temperature case were selected as the bounding analysis conditions. The modeled power level of 3579 MWth adjusted for calorimetric error (+2% of power) was the basis in the analysis. The use of higher temperatures is conservative because the initial fluid energy is based on coolant temperatures which are at the maximum levels attained in steady state operation. Additionally, an allowance of +4.85°F is reflected in the temperatures in order to account for instrument error and deadband. The initial RCS pressure in this analysis is based on a nominal value of 2250 psia. Also included is an allowance of +50 psia, which accounts for the uncertainty on pressurizer pressure. The selection of 2250 psia as the limiting pressure is considered to affect the blowdown phase results only since this represents the initial pressure of the RCS. The RCS rapidly depressurizes from this value until the point at which it equilibrates with containment pressure.

The rate at which the RCS blows down is initially more severe at the higher RCS pressure (2250 psia). Additionally the RCS has a higher fluid density at 2250 psia (assuming a constant temperature) and subsequently has a higher RCS mass available for release. Thus, 2250 psia initial pressure was selected as the limiting case for the long term mass and energy release calculations. These assumptions conservatively maximize the mass and energy in the RCS.

The selection of fuel allowance for the long term mass and energy calculation and subsequent LOCA containment integrity calculation is based on the need to conservatively maximize the core stored energy. The margin in core stored energy was chosen to be +15%. Thus, the analysis very conservatively accounts for the stored energy in the core.

Margin in RCS volume of 3% (which is composed of 1.6% allowance for thermal expansion, and 1.4% for uncertainty) is modeled.

Regarding safety injection flow, the mass and energy calculation considers both minimum and maximum safety injection flowrates.

3.4.2.3 LOCA Mass and Energy Release Phases

The LOCA transient is typically divided into four phases:

1. Blowdown - which includes the period from accident initiation (when the reactor is at steady state operation) to the time that the RCS pressure reaches initial equilibrium with containment.

2. Refill - the period of time when the lower plenum is being filled by accumulator and safety injection water. At the end of blowdown, a large amount of water remains in the cold legs, downcomer, and lower plenum to conservatively consider the refill period for the purpose of containment mass and energy releases, this water is instantaneously transferred to the lower plenum along with sufficient accumulator water to completely fill the lower plenum. This allows an uninterrupted release of mass and energy to containment. Thus, the refill period is conservatively neglected in the mass and energy release calculation.

3. Reflood - begins when the water from the lower plenum enters the core and ends when the core is completely quenched.

4. Post-Reflood (Froth) - describes the period following the reflood transient. For the pump suction break, a two-phase mixture exits the core, passes through the hot legs, and is superheated in the steam generators. After the broken loop steam generator cools, the break flow becomes two phase.

3.4.2.4 Break Size and Location

Generic studies have been performed with respect to the effect on the LOCA mass and energy releases relative to postulated break size. The double ended guillotine break has been found to be limiting due to larger mass flow rates during the blowdown phase of the transient. During the reflood and froth phases, the break size has little effect on the releases.

Three distinct locations in the reactor coolant system loop can be postulated for pipe rupture:

1. Hot leg (between vessel and steam generator)
2. Cold leg (between pump and vessel)
3. Pump suction (between steam generator and pump)

The break location analyzed and described herein is the double-ended pump suction guillotine break (10.48 ft^2). Pump suction break mass and energy releases have been calculated for the blowdown, reflood, and post-reflood phases of the LOCA.

The following information provides a discussion on each break location. The double ended hot leg guillotine has been shown in previous studies to result in the highest blowdown mass and energy release rates. Although the core flooding rate would be highest for this break location, the amount of energy released from the steam generator secondary side is minimal because the majority of the fluid which exits the core bypasses the steam generators in venting to containment. As a result, the reflood mass and energy releases are reduced significantly as compared to either the pump suction or cold leg break locations where the core exit mixture must pass through the steam generators before venting through the break.

For the hot leg break, there is no reflood peak as determined by generic studies (i.e., from the end of the blowdown period the releases would continually decrease). Therefore the reflood (and subsequent post-reflood) releases are not calculated for a hot leg break. The mass and energy releases for the hot leg break blowdown phase have been included in the scope of this containment integrity analysis.

The cold leg break location has also been found in previous studies to be much less limiting in terms of the overall containment peak pressure. The cold leg blowdown is faster than that of the pump suction break, and more mass is released into the containment. However, the core heat transfer is greatly reduced, and this results in a considerably lower energy release into containment. Studies have determined that the blowdown transient for the cold leg is, in general, less limiting than that for the pump suction break. During reflood, the flooding rate is greatly reduced and the energy release rate into the containment is reduced. Therefore, the cold leg break is not usually performed.

The pump suction break combines the effects of the relatively high core flooding rate, as in the hot leg break, and the addition of the stored energy in the steam generators.

As a result, the pump suction break yields the highest energy flow rates during the post-blowdown period by including all of the available energy of the Reactor Coolant System in calculating the releases to containment. This break location has been determined to be the limiting break for typical dry containment plants. The choice of this break location for Wolf Creek as the limiting break is consistent with other dry containment plants for the post blowdown phase of the event.

In summary then, the analysis of the limiting break location for a dry containment has been performed and is shown in this report. The double-ended pump suction guillotine break has historically been considered to be the limiting break location for the post blowdown phase of the event, by virtue of its consideration of all energy sources present in the RCS. The analyses presented in this document support the conclusions of the double ended pump suction (DEPS) as the limiting break case for the post blowdown period, considering both the minimum and maximum safety injection cases. This break location provides a mechanism for the release of the available energy in the Reactor Coolant System, including both the broken and intact loop steam generators.

3.4.2.5 Application of Single Failure Criteria

An analysis of the effects of the single failure criteria has been performed on the mass and energy release rates for the (DEPS) break. For the DEPS results presented in this report, an inherent assumption in the generation of the mass and energy release is that offsite power is lost. This results in the actuation of the emergency diesel generators, required to power the safety injection system. This is not an issue for the blowdown period which is limited by the double ended hot leg (DEHL) break.

Two cases have been analyzed for the effects of a single failure. The double ended pump suction case with both minimum and maximum safety injection for the 3579 MWth rerated conditions was analyzed. The limiting case for Wolf Creek is the minimum safeguards case. This was determined by prior generic and specific Wolf Creek analyses. In the case of minimum safeguards, the single failure postulated to occur is the loss of an emergency diesel generator.

This results in the loss of one pumped safety injection train and the containment safeguards components on that diesel, thereby minimizing the safety injection flow. The analysis further considers the safety injection pump head curves to be degraded by 10%. This results in the greatest reduction possible for the Emergency Core Cooling System (ECCS) components. For the case analyzing maximum safety injection, a conservative assumption was made due to the availability of only the diesel train failure criteria. The maximum safety injection flows were modeled assuming that the containment analysis is performed when the minimum containment safeguard components are available. This applies to heat exchangers, fan coolers and the containment spray system. This assumption would provide a bounding assessment in maximizing the mass release but minimizing the heat removal capability.

The following items ensure that the mass and energy releases are conservatively calculated for maximum containment pressure:

1. Maximum expected operating temperature of the reactor coolant system

2. Allowance in temperature for instrument error and dead band (+4.85°F)

3. Margin in volume of 3% (which is composed of 1.6% allowance for thermal expansion, and 1.4% for uncertainty)

4. Power level of 3579 MWth

5. Allowance for calorimetric error (+2% of power)

6. Conservative coefficients of heat transfer (i.e., steam generator primary/secondary heat transfer and reactor coolant system metal heat transfer)

7. Allowance in core stored energy effect of fuel densification

8. Margin in core stored energy (+15%)

9. Allowance for RCS pressure uncertainty (+50 psi)

3.4.2.6 Blowdown Mass and Energy Release Data

The SATAN-VI code is used for computing the blowdown transient, and is the same as that used for the ECCS calculation in Reference 2. The methodology for the use of this model is described in Reference 1.

Tables 3.4.2-1 and 3.4.2-2 present the calculated mass and energy releases for the blowdown phase of the break analyzed for the DEPS and DEHL breaks, respectively. Break flow time histories from each side of the guillotine break are tabulated, where Break Flow Path No. 1 represents the flow from the reactor vessel outlet side of the break and Break Flow Path No. 2 represents the flow from the reactor vessel inlet side of the break. The mass and energy release for the double-ended pump suction break and the double-ended hot leg break, given in 3.4.2-1 and 3.4.2-2 terminate 22.3 and 25.5 seconds respectively after the initiation of the postulated accident.

3.4.2.7 Reflood Mass and Energy Release Data

The WREFLOOD code is used for computing the reflood transient, and is a modified version of that used in the ECCS calculation in Reference 2. The methodology for the use of this model is described in Reference 1.

An exception to the mass and energy evaluation model described in Reference 1 is taken, in that steam/water mixing in the broken loop has been included in this analysis. This assumption is justified and is supported by test data, and is summarized as follows:

The model assumes a complete mixing condition (i.e., thermal equilibrium) for the steam/water interaction. The complete mixing process, however, is made up of two distinct physical processes. The first is a two phase interaction with condensation of steam by cold injection water. The second is a single phase mixing of condensate and injection water. Since the mass and energy of the steam released is the most important influence to the containment pressure transient, the steam condensation part of the mixing process is the only part that need be considered. (Any spillage directly heats only the sump.)

The most applicable steam/water mixing test data has been reviewed for validation of the containment integrity reflood steam/water mixing model. This data is that generated in 1/3 scale test (Reference 3), which are the largest scale data available and thus most closely simulates the flow regimes and gravitational effects that would occur in a PWR. These tests were designed specifically to study the steam/water interaction for PWR reflood conditions.

From the entire series of 1/3 scale test, a group corresponds almost directly to containment integrity reflood conditions. The injection flow rates for this group cover all phases and mixing conditions calculated during the reflood transient. The data from these tests were reviewed and discussed in detail in Reference 1. For all of these tests, the data clearly indicates the occurrence of very effective mixing with rapid steam condensation. The mixing model used in the containment integrity reflood calculation is therefore wholly supported by the 1/3 scale steam/water mixing data.

Additionally, the following justification is also noted. The limiting break for the containment integrity peak pressure analysis during the post-blowdown phase is the double ended pump suction break. For this break, there are two flow paths available in the RCS by which mass and energy may be released to containment. One is through the outlet of the steam generator, the other via reverse flow through the reactor coolant pump. Steam, which is not condensed by ECC injection in the intact RCS loops passes around the downcomer and through the broken loop cold leg and pump in venting to containment. This steam also encounters ECC injection water as it passes through the broken loop cold leg, complete mixing occurs and a portion of it is condensed. It is this portion of steam which is condensed that is taken credit for in this analysis. This assumption is justified based upon the postulated break location, and the actual physical presence of the ECC injection nozzle. A description of the test and test results is contained in References 1 and 3.

The methodology previously discussed and described in Reference 1 has been utilized and approved on the Dockets for Catawba Units 1 and 2, Indian Point 2 and 3, McGuire Units 1 and 2, Sequoyah Units 1 and 2, Watts Bar Units 1 and 2, Millstone Unit 3, Beaver Valley Unit 2, Surry Units 1 and 2, and Vogtle Units 1 and 2.

Tables 3.4.2-3 and 3.4.2-4 present the calculated mass and energy release for the reflood phase of the Double-Ended Pump Suction break, with minimum and maximum safety injection respectively. Flow time histories from each side of the Double-Ended Pump Suction break are tabulated, where Break Flow Path No. 1 represents the flow through the outlet of the steam generator and Break Flow Path No. 2 represents reverse flow through the reactor coolant pump. A significantly higher mass and energy release occurs during the period the accumulators are injecting (from 28.9 to 49.9 seconds for minimum and maximum safety injection as illustrated in Table 3.4.2-3 and 3.4.2-4).

The transient of the principal parameters during reflood are given in Tables 3.4.2-5 and 3.4.2-6 for the minimum and maximum safety injection double ended pump suction break cases.

3.4.2.8 Post-Reflood Mass and Energy Release Data

The FROTH code (Reference 4) is used for computing the post-reflood transient. The methodology for the use of this model is described in Reference 1. The mass and energy release rates calculated by FROTH are used in the containment analysis until the time of containment depressurization.

After depressurization, the mass and energy release from decay heat is based on the 1979 ANSI/ANS Standard, shown in Reference 5, and the following input:

1. Decay heat sources considered are fission product decay and heavy element decay of U-239 and Np-239.

2. Decay heat power from fissioning isotopes other than U-235 is assumed to be identical to that of U-235.

3. Fission rate is constant over the operating history of maximum power level.

4. The factor accounting for neutron capture in fission products has been taken from Table 10 of ANS (1979).

5. Operation time before shutdown is 3 years.

6. The total recoverable energy associated with one fission has been assumed to be 200 MeV/fission.

7. Two sigma uncertainty (2 times the standard deviation) has been applied to the fission product decay.

Tables 3.4.2-7 and 3.4.2-8 present the two phase (froth) mass and energy release data for the Double-Ended Pump Suction break minimum and maximum safety injection cases. Flow time histories from each side of the Double-Ended Pump Suction break are tabulated, where Break Flow Path No. 1 represents the flow through the outlet of the steam generator and Break Flow Path No. 2 represents reverse flow through the reactor coolant pump.

3.4.2.9 Sources of Mass and Energy

The sources of mass considered in the LOCA mass and energy release analysis are given in Tables 3.4.2-9, 3.4.2-10, and 3.4.2-11. These sources are the reactor coolant system, accumulators, and pumped safety injection.

The energy inventories considered in the LOCA mass and energy release analysis are given in Tables 3.4.2-12, 3.4.2-13, and 3.4.2-14. The energy sources include:

1. Reactor Coolant System Water
2. Accumulator Water
3. Pumped Injection Water
4. Decay Heat
5. Core Stored Energy
6. Reactor Coolant System Metal
7. Steam Generator Metal
8. Steam Generator Secondary Energy
9. Secondary Transfer of Energy (feedwater into and steam out of the steam generator secondary)

In the mass and energy release data presented, no Zirc-water reaction heat was considered because the clad temperature did not rise high enough for the rate of the Zirc-water reaction heat to be of any significance.

System parameters needed to perform confirmatory analyses are provided in Table 3.4.2-15.

The consideration of the various energy sources in the mass and energy release analysis provides assurance that all available sources of energy have been included in this analysis. Thus the review guidelines presented in Standard Review Plan Section 6.2.1.3 have been satisfied.

The mass and energy inventories are presented at the following times, as appropriate:

1. Time zero (initial conditions)
2. End of blowdown time
3. End of refill time
4. End of reflood time
5. Time of full depressurizations
6. End of analysis

The methods and assumptions used to release the various energy sources are given in Reference 1, except as noted in section 3.4.2.7, which has been approved as a valid evaluation model by the Nuclear Regulatory Commission.

TABLE 3.4.2-1

BLOWDOWN MASS AND ENERGY RELEASES
DOUBLE-ENDED PUMP SUCTION

TIME	BREAK PATH NO.1 FLOW		BREAK PATH NO.2 FLOW	
		THOUSAND		THOUSAND
SECONDS	LBM/SEC	BTU/SEC	LBM/SEC	BTU/SEC
0.0000	0.0	0.0	0.0	0.0
0.0501	42274.0	23548.4	23902.6	13239.3
0.100	42317.3	23630.3	22280.5	12386.3
0.250	43583.8	24646.6	24672.4	13735.0
0.450	45254.5	26199.6	23305.4	13003.9
0.651	44726.2	26498.4	20949.7	11703.3
1.00	40236.5	24553.7	18816.9	10538.5
1.20	38253.9	23652.0	18433.5	10333.5
1.90	33325.7	21723.5	17808.4	9989.8
2.20	30553.0	20426.9	17654.6	9911.2
2.60	26276.6	18100.3	17104.3	9616.0
2.80	21962.8	15304.0	16529.9	9299.8
3.80	17205.3	12198.7	14625.0	8258.6
4.20	15670.4	11120.3	14157.6	8001.6
5.00	14090.4	9964.4	13353.8	7549.1
5.50	13515.3	9514.4	14273.9	8066.1
6.75	13023.5	9066.4	13531.1	7640.3
7.75	12258.5	8562.1	13048.7	7372.9
8.50	12017.6	8365.5	12628.1	7139.9
8.75	11636.4	8222.1	12617.1	7136.3
9.00	10194.4	7760.2	12350.3	6985.2
9.25	9564.9	7467.9	12234.3	6921.4
10.5	9235.0	6977.8	11328.7	6413.7
11.5	8753.8	6505.9	10682.5	6046.4
12.8	7750.2	5958.5	9862.8	5610.9
13.8	6526.0	5512.3	9153.8	5225.1
15.3	5412.5	4967.4	8143.1	4665.0
16.0	4859.3	4720.8	7399.7	4254.1
17.0	4025.2	4292.9	6920.6	3632.6
19.3	1612.2	2022.7	3895.8	1590.2
19.5	1376.6	1733.7	2940.3	1210.3
20.0	1050.5	1330.6	2554.0	998.1
20.5	795.9	1009.8	1842.6	701.1
21.3	485.6	617.9	1275.3	533.3
22.3	124.6	159.2	266.1	203.1

TABLE 3.4.2-2

BLOWDOWN MASS AND ENERGY RELEASES
DOUBLE-ENDED HOT LEG GUILLOTINE

TIME	BREAK PATH NO.1 FLOW		BREAK PATH NO.2 FLOW	
SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
0.0000	0.0	0.0	0.0	0.0
0.0502	39452.8	25936.9	28757.0	18595.4
0.101	43473.3	28496.2	28172.5	18214.3
0.251	36632.5	23945.3	22454.4	14332.0
0.350	35539.3	23203.6	20813.2	13076.4
0.650	34530.0	22564.2	18605.9	11170.2
0.951	32408.8	21459.7	17736.8	10344.5
1.20	31725.8	21287.8	17624.2	10116.2
1.40	30303.0	20543.1	17762.0	10093.7
1.90	28103.3	19339.1	18235.1	10202.8
2.40	25934.6	18001.1	18240.2	10148.3
2.90	24168.8	16822.7	17913.8	9968.0
3.50	22605.7	15683.6	17246.9	9638.8
4.20	21431.4	14741.4	16192.1	9117.1
4.70	20878.9	14268.5	15023.4	8510.7
6.25	20889.3	13923.0	12146.8	6976.3
7.00	21194.0	13931.4	11214.3	6459.6
7.25	15218.8	10927.4	10929.3	6300.7
7.50	16501.0	11675.9	10657.0	6148.9
8.00	16376.9	11535.0	10158.2	5870.3
9.75	15659.1	10661.3	8337.6	4873.9
10.3	15183.5	10277.4	7805.1	4590.7
12.3	12155.9	8486.9	5880.0	3597.6
14.0	9354.9	6963.6	4456.5	2916.5
14.8	7645.7	6178.9	3809.1	2614.3
15.5	5577.7	5300.5	3008.9	2240.0
16.5	3893.4	4010.8	2381.7	1884.4
17.3	2265.0	2671.3	1998.5	1702.3
18.8	1207.9	1498.8	852.7	1052.8
20.3	479.1	608.2	264.5	332.1
21.3	155.7	196.7	0.0	0.0
22.5	59.5	76.4	127.4	163.5
23.0	230.6	299.0	0.0	0.0
24.3	776.7	922.9	211.5	267.8
25.5	0.0	0.0	0.0	0.0

TABLE 3.4.2-3

REFLOOD MASS AND ENERGY RELEASES
DOUBLE-ENDED PUMP SUCTION - MIN SI

TIME	BREAK PATH NO.1 FLOW		BREAK PATH NO.2 FLOW	
SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
22.8	0.0	0.0	0.0	0.0
23.7	0.0	0.0	0.0	0.0
23.9	56.3	66.5	0.0	0.0
24.0	31.4	37.1	0.0	0.0
26.8	139.3	164.7	0.0	0.0
27.8	163.2	193.0	0.0	0.0
28 9	528.6	628.2	4944.8	716.5
29.9	541.6	644.0	5030.9	751.3
30.9	531.4	631.7	4944.0	742.3
32.9	509.3	605.2	4751.0	720.5
33.0	508.2	603.9	4741.3	719.4
34.9	488.1	579.8	4561.0	698.3
36.9	468.5	556.4	4382.0	677.3
38.9	450.5	534.9	4215.1	657.6
40.9	434.1	515.3	4059.6	639.3
42.9	419.0	497.3	3914.5	622.1
43.9	447.0	530.7	4205.7	640.2
47.9	421.8	500.6	3965.5	610.8
49.9	410.3	486.9	3853.9	597.3
50.9	398.3	472.3	311.3	218.9
51.1	435.1	516.4	327.8	241.3
51.9	459.6	545.8	338.5	256.5
52.9	449.2	533.3	333.7	250 1
57.9	394.0	467.4	308.4	216.7
64.9	333.4	395.1	280.8	180.9
71.9	285.6	338.3	259.4	153.5
79.9	244.0	288.8	241.2	130.3
91.9	202.3	239.3	223.4	107.8
95.9	192.8	228.1	219.4	102.9
107.9	173.8	205.6	211.6	93.0
121.9	163.8	193.7	207.4	87.8
135.9	160.8	190.2	206.0	86.1
147.9	161.0	190.4	205.9	85.9
191.9	168.1	198.9	208.0	88.6
196.8	169.1	200.1	208.4	89.0

TABLE 3.4.2-4

REFLOOD MASS AND ENERGY RELEASES
DOUBLE-ENDED PUMP SUCTION - MAX SI

TIME	BREAK PATH NO.1 FLOW		BREAK PATH NO.2 FLOW	
SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
22.8	0.0	0.0	0.0	0.0
23.7	0.0	0.0	0.0	0.0
23.9	56.3	66.5	0.0	0.0
24.0	31.4	37.1	0.0	0.0
26.8	139.3	164.7	0.0	0.0
27.8	163.2	193.0	0.0	0.0
28.9	528.6	628.2	4944.8	716.5
29.9	541.6	644.0	5030.9	751.3
30.9	531.4	631.7	4944.0	742.3
34.9	488.1	579.8	4561.0	698.3
36.9	468.5	556.4	4382.0	677.3
38.9	450.5	534.9	4215.1	657.6
40.9	434.1	515.3	4059.6	639.3
42.9	419.0	497.3	3914.5	622.1
43.9	460.6	547.0	4343.1	650.1
47.9	435.9	517.4	4109.8	621.1
49.9	424.5	503.8	4000.0	607.9
50.9	362.7	430.0	339.1	189.9
51.9	380.6	451.4	346.1	200.1
61.9	328.2	389.0	324.2	170.7
66.1	311.5	369.1	317.4	161.6
75.9	282.9	335.1	305.9	146.2
87.9	260.4	308.3	296.9	134.4
99.9	247.7	293.2	291.9	127.8
105.9	246.3	291.6	293.7	127.4
113.9	247.1	292.5	305.4	129.1
121.9	248.0	293.6	323.3	132.0
129.9	246.7	292.1	344.0	134.7
137.9	242.4	286.9	366.4	136.9
145.9	234.7	277.8	390.0	138.6
151.9	226.7	268.3	408.4	139.7
159.9	213.2	252.3	434.4	141.2
167.9	196.7	232.7	462.6	143.3
169.9	192.0	227.2	470.2	143.9
173.8	182.4	215.8	485.5	145.3

TABLE 3.4.2-5
PRINCIPAL PARAMETERS DURING REFLOOD
DOUBLE-ENDED PUMP SUCTION - MINIMUM SI

TIME	FLOODING TEMP	FLOODING RATE	CARRYOVER FRACTION	CORE HEIGHT	DOWNCOMER HEIGHT	FLOW FRACTION	INJECTION TOTAL	INJECTION ACCUMULATOR	INJECTION SPILL	ENTHALPY
SECONDS	DEGREE F	IN/SEC		FT	FT		(POUNDS MASS PER SECOND)			BTU/LBM
22.8	260.7	0.000	0.000	0.00	0.00	0.250	0.0	0.0	0.0	0.00
23.5	255.6	22.123	0.000	0.55	1.70	0.000	8363.9	8363.9	0.0	89.60
23.7	251.4	26.295	0.000	1.06	1.65	0.000	8267.1	8267.1	0.0	89.60
24.8	248.4	2.887	0.313	1.50	4.86	0.322	7908.9	7908.9	0.0	89.60
25.7	247.6	2.784	0.436	1.63	7.77	0.345	7672.6	7672.6	0.0	89.60
28.9	244.7	5.286	0.647	2.00	16.05	0.609	6404.0	6404.0	0.0	89.60
30.9	242.0	4.872	0.702	2.27	16.07	0.603	5999.8	5999.8	0.0	89.60
33.0	239.8	4.551	0.726	2.51	16.07	0.598	5699.6	5699.6	0.0	89.60
38.3	236.1	4.068	0.748	3.00	16.07	0.584	5089.1	5089.1	0.0	89.60
42.9	234.3	3.793	0.753	3.37	16.07	0.571	4668.7	4668.7	0.0	89.60
43.9	233.9	3.949	0.755	3.45	16.07	0.584	5001.6	4482.8	0.0	86.84
44.5	233.8	3.921	0.755	3.50	16.07	0.583	4959.5	4439.5	0.0	86.81
49.9	232.6	3.692	0.757	3.92	16.07	0.572	4585.3	4057.7	0.0	86.54
50.9	232.5	3.667	0.758	3.99	16.03	0.586	534.3	0.0	0.0	63.01
51.1	232.5	3.849	0.759	4.00	15.98	0.592	523.9	0.0	0.0	63.01
51.9	232.4	3.957	0.759	4.07	15.75	0.596	515.2	0.0	0.0	63.01
57.9	233.0	3.455	0.757	4.51	14.29	0.586	529.3	0.0	0.0	63.01
65.4	235.9	2.970	0.754	5.00	12.98	0.573	540.7	0.0	0.0	63.01
74.9	241.5	2.515	0.750	5.54	11.96	0.554	550.0	0.0	0.0	63.01
84.4	248.1	2.197	0.748	6.00	11.45	0.535	555.6	0.0	0.0	63.01
95.9	255.8	1.952	0.747	6.50	11.27	0.516	559.4	0.0	0.0	63.01
108.5	262.8	1.803	0.747	7.00	11.42	0.501	561.4	0.0	0.0	63.01
123.9	269.8	1.719	0.750	7.57	11.86	0.493	562.5	0.0	0.0	63.01
136.2	274.7	1.693	0.754	8.00	12.29	0.491	562.7	0.0	0.0	63.01
151.9	280.0	1.684	0.760	8.54	12.89	0.492	562.7	0.0	0.0	63.01
155.9	281.3	1.684	0.761	8.67	13.04	0.493	562.7	0.0	0.0	63.01
165.8	284.2	1.685	0.765	9.00	13.42	0.494	562.6	0.0	0.0	63.01
181.9	288.4	1.691	0.771	9.53	14.05	0.497	562.4	0.0	0.0	63.01
196.8	291.8	1.698	0.777	10.00	14.62	0.500	562.2	0.0	0.0	63.01

TABLE 3.4.2-6

PRINCIPAL PARAMETERS DURING REFLOOD
DOUBLE-ENDED PUMP SUCTION - MAXIMUM SI

TIME	FLOODING		CARRYOVER	CORE	DOWNCOMER	FLOW		INJECTION		
	TEMP	RATE	FRACTION	HEIGHT	HEIGHT	FRACTION	TOTAL	ACCUMULATOR	SPILL	ENTHALPY
SECONDS	DEGREE F	IN/SEC		FT	FT		(POUNDS MASS PER SECOND)			BTU/LBM
22.8	260.7	0.000	0.000	0.00	0.00	0.250	0.0	0.0	0.0	0.00
23.5	255.6	22.123	0.000	0.55	1.70	0.000	8363.9	8363.9	0.0	89.60
23.7	251.4	26.295	0.000	1.06	1.65	0.000	8267.1	8267.1	0.0	89.60
24.8	248.4	2.887	0.313	1.50	4.86	0.322	7908.9	7908.9	0.0	89.60
25.7	247.6	2.784	0.436	1.63	7.77	0.345	7672.6	7672.6	0.0	89.60
28.9	244.7	5.286	0.647	2.00	16.05	0.609	6404.0	6404.0	0.0	89.60
30.9	242.0	4.872	0.702	2.27	16.07	0.603	5999.8	5999.8	0.0	89.60
33.0	239.8	4.551	0.726	2.51	16.07	0.598	5699.6	5699.6	0.0	89.60
38.3	236.1	4.068	0.748	3.00	16.07	0.584	5089.1	5089.1	0.0	89.60
42.9	234.3	3.793	0.753	3.37	16.07	0.571	4668.7	4668.7	0.0	89.60
43.9	233.9	4.031	0.755	3.45	16.07	0.589	5160.5	4438.6	0.0	85.88
44.5	233.8	4.003	0.756	3.50	16.07	0.589	5121.9	4398.0	0.0	85.84
49.9	232.5	3.774	0.758	3.93	16.07	0.578	4753.0	4018.4	0.0	85.49
50.9	232.4	3.435	0.757	4.00	16.04	0.552	754.4	0.0	0.0	63.01
51.9	232.4	3.513	0.757	4.07	15.93	0.557	747.3	0.0	0.0	63.01
58.9	233.5	3.222	0.757	4.55	15.29	0.547	756.6	0.0	0.0	63.01
66.1	236.7	2.991	0.757	5.00	14.90	0.536	763.5	0.0	0.0	63.01
74.9	242.3	2.792	0.757	5.52	14.69	0.526	769.0	0.0	0.0	63.01
83.8	249.1	2.648	0.759	6.00	14.69	0.518	772.7	0.0	0.0	63.01
93.9	256.9	2.536	0.762	6.53	14.89	0.511	775.3	0.0	0.0	63.01
103.6	263.4	2.477	0.765	7.00	15.19	0.507	776.5	0.0	0.0	63.01
115.9	270.5	2.452	0.770	7.59	15.59	0.511	776.3	0.0	0.0	63.01
124.8	275.0	2.429	0.773	8.00	15.80	0.515	776.0	0.0	0.0	63.01
135.9	279.9	2.368	0.777	8.50	15.97	0.517	776.6	0.0	0.0	63.01
147.5	284.2	2.264	0.780	9.00	16.04	0.514	778.3	0.0	0.0	63.01
161.9	288.7	2.084	0.782	9.57	16.07	0.501	782.1	0.0	0.0	63.01
173.8	291.8	1.898	0.784	10.00	16.07	0.478	786.2	0.0	0.0	63.01

TABLE 3.4.2-7

POST REFLOOD MASS AND ENERGY RELEASES
DOUBLE-ENDED PUMP SUCTION - MIN SI

TIME	BREAK PATH NO.1 FLOW		BREAK PATH NO.2 FLOW	
		THOUSAND		THOUSAND
SECONDS	LBM/SEC	BTU/SEC	LBM/SEC	BTU/SEC
196.8	215.6	270.3	348.6	130.6
201.8	214.5	269.0	349.6	130.7
206.8	214.7	269.2	349.4	130.4
236.8	211.3	264.9	352.8	129.9
241.8	211.3	265.0	352.8	129.6
276.8	207.4	260.1	356.7	129.0
286.8	207.3	259.9	356.9	128.6
301.8	204.9	257.0	359.2	128.5
311.8	204.6	256.5	359.5	128.1
326.8	202.9	254.4	361.2	127.8
376.8	196.7	246.6	367.4	127.0
381.8	196.8	246.8	367.3	126.7
396.8	194.6	244.0	369.6	126.5
401.8	203.9	255.7	360.2	129.0
406.8	203.6	255.3	360.5	128.8
451.8	198.2	248.6	365.9	127.9
461.8	197.5	247.6	366.6	127.5
476.8	195.2	244.8	368.9	127.3
481.8	195.0	244.5	369.1	127.1
506.8	191.7	240.4	372.4	126.6
686.8	191.7	240.4	372.4	126.6
686.9	98.4	122.4	465.7	147.2
801.8	95.2	118.4	468.9	144.3
811.8	95.0	118.1	469.1	148.0
951.8	91.9	114.2	472.2	144.2
1056.8	89.9	111.6	474.2	143.4
1106.8	89.1	110.6	475.1	143.0
1301.8	85.9	106.6	478.2	139.2
1426.7	85.7	106.4	478.4	139.9
1426.8	82.1	94.5	482.0	43.8
1508.9	82.1	94.5	482.0	43.8
1509.0	93.4	107.6	443.7	127.8
3599.9	93.4	107.6	443.7	127.8
3600.0	65.2	75.0	471.9	135.9
3600.1	58.1	66.9	478.9	111.1
10000.0	42.3	48.6	494.8	114.8
100000.0	22.6	26.0	514.5	119.4
1000000.0	9.7	11.1	527.4	122.3

TABLE 3.4.2-8

POST REFLOOD MASS AND ENERGY RELEASES
DOUBLE-ENDED PUMP SUCTION - MAX SI

TIME	BREAK PATH NO.1 FLOW		BREAK PATH NO.2 FLOW	
		THOUSAND		THOUSAND
SECONDS	LBM/SEC	BTU/SEC	LBM/SEC	BTU/SEC
173.9	148.2	184.9	640.6	171.1
183.9	148.7	185.5	640.1	170.4
228.9	146.7	183.1	642.1	168.5
243.9	147.3	183.8	641.5	167.5
258.9	145.9	182.1	642.8	167.0
273.9	146.4	182.8	642.3	166.1
328.9	144.0	179.8	644.7	163.6
343.9	144.4	180.2	644.4	162.7
368.9	142.8	178.2	646.0	161.7
403.9	141.9	177.1	646.8	159.8
423.9	142.1	177.4	646.6	158.6
448.9	141.0	176.0	647.8	157.5
463.9	141.4	176.5	647.3	156.5
473.9	140.6	175.5	648.2	156.1
508.9	140.7	175.6	648.0	158.8
518.9	139.8	174.5	649.0	158.4
538.9	140.2	175.0	648.6	157.0
563.9	139.1	173.6	649.7	155.6
588.9	139.2	173.7	649.6	153.9
608.9	138.3	172.6	650.5	152.8
633.9	138.5	172.9	650.2	155.6
658.9	137.6	171.7	651.2	154.0
663.9	138.0	172.3	650.7	153.5
773.9	136.0	169.7	652.8	149.4
1108.9	136.0	169.7	652.8	149.4
1109.0	90.7	112.1	669.5	254.1
1118.9	90.6	111.9	669.7	254.6
1233.9	88.7	109.6	671.5	251.4
1346.8	88.7	109.6	671.5	251.4
1346.9	85.1	97.9	675.1	152.3
3600.0	66.9	77.0	693.3	155.6
3600.1	55.7	64.0	704.6	134.8
10000.0	40.5	46.6	719.8	137.7
100000.0	21.6	24.9	738.6	141.3
1000000.0	9.3	10.7	751.0	143.6

TABLE 3.4.2-9

MASS BALANCE
DOUBLE-ENDED PUMP SUCTION - MIN SI

	TIME (SECONDS)	0.00	22.25	22.25	196.75	691.80	1426.67	3600.00
				MASS (THOUSAND LBM)				
INITIAL	IN RCS AND ACC	711.99	711.99	711.99	711.99	711.99	711.99	711.99
ADDED MASS	PUMPED INJECTION	0.00	0.00	0.00	85.14	364.38	775.95	1943.16
	TOTAL ADDED	0.00	0.00	0.00	85.14	364.38	775.95	1943.16
	*** TOTAL AVAILABLE ***	711.99	711.99	711.99	797.13	1076.37	1487.94	2655.15
DISTRIBUTION	REACTOR COOLANT	501.71	48.02	61.92	120.30	120.30	120.30	120.30
	ACCUMULATOR	210.28	172.23	158.32	0.00	0.00	0.00	0.00
	TOTAL CONTENTS	711.99	220.24	220.24	120.30	120.30	120.30	120.30
EFFLUENT	BREAK FLOW	0.00	491.74	491.74	661.38	940.61	1352.19	2519.40
	ECCS SPILL	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	TOTAL EFFLUENT	0.00	491.74	491.74	661.38	940.61	1352.19	2519.40
	*** TOTAL ACCOUNTABLE ***	711.99	711.99	711.99	781.67	1060.91	1472.48	2639.69

TABLE 3.4.2-10

MASS BALANCE
DOUBLE-ENDED PUMP SUCTION - MAX SI

	TIME (SECONDS)	0.00	22.25	22.25	173.80	1113.90	1346.82	3600.00
				MASS (THOUSAND LBM)				
INITIAL	IN RCS AND ACC	711.99	711.99	711.99	711.99	711.99	711.99	711.99
ADDED MASS	PUMPED INJECTION	0.00	0.00	0.00	100.55	834.57	1011.65	2724.61
	TOTAL ADDED	0.00	0.00	0.00	100.55	834.57	1011.65	2724.61
	*** TOTAL AVAILABLE ***	711.99	711.99	711.99	812.54	1546.56	1723.64	3436.60
DISTRIBUTION	REACTOR COOLANT	501.71	48.02	61.92	122.94	122.94	122.94	122.94
	ACCUMULATOR	210.28	172.23	158.32	0.00	0.00	0.00	0.00
	TOTAL CONTENTS	711.99	220.24	220.24	122.94	122.94	122.94	122.94
EFFLUENT	BREAK FLOW	0.00	491.74	491.74	674.14	1408.17	1585.24	3298.20
	ECCS SPILL	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	TOTAL EFFLUENT	0.00	491.74	491.74	674.14	1408.17	1585.24	3298.20
	*** TOTAL ACCOUNTABLE ***	711.99	711.99	711.99	797.08	1531.11	1708.18	3421.14

TABLE 3.4.2-11

MASS BALANCE
DOUBLE-ENDED HOT LEG GUILLOTINE

	TIME (SECONDS)	0.00	25.50	25.50
		MASS (THOUSAND LBM)		
INITIAL	IN RCS AND ACC	711.99	711.99	711.99
ADDED MASS	PUMPED INJECTION	0.00	0.00	0.00
	TOTAL ADDED	0.00	0.00	0.00
	*** TOTAL AVAILABLE ***	711.99	711.99	711.99
DISTRIBUTION	REACTOR COOLANT	501.71	99.02	112.92
	ACCUMULATOR	210.28	130.78	116.88
	TOTAL CONTENTS	711.99	229.80	229.80
EFFLUENT	BREAK FLOW	0.00	482.18	482.18
	ECCS SPILL	0.00	0.00	0.00
	TOTAL EFFLUENT	0.00	482.18	482.18
	*** TOTAL ACCOUNTABLE ***	711.99	711.99	711.99

TABLE 3.4.2-12

ENERGY BALANCE: DOUBLE-ENDED PUMP SUCTION - MIN SI

	TIME (SECONDS)	0.00	22.25	22.25	196.75	691.80	1426.67	3600.00
				ENERGY (MILLION BTU)				
INITIAL	IN RCS/ACC/S GEN	883.61	883.61	883.61	883.61	883.61	883.61	883.61
ADDED ENERGY	PUMPED INJECTION	0.00	0.00	0.00	5.36	22.96	58.86	329.64
	DECAY HEAT	0.00	8.49	8.49	30.21	76.55	132.58	263.01
	HEAT FROM SECONDARY	0.00	-1.71	-1.71	-1.71	-0.80	0.34	0.34
	TOTAL ADDED	0.00	6.78	6.78	33.87	98.71	191.79	592.99
	*** TOTAL AVAILABLE ***	883.61	890.40	890.40	917.48	982.32	1075.40	1476.61
DISTRIBUTION	REACTOR COOLANT	302.21	15.12	16.36	34.87	34.87	34.87	34.87
	ACCUMULATOR	18.82	15.41	14.17	0.00	0.00	0.00	0.00
	CORE STORED	27.84	13.40	13.40	5.09	4.91	4.51	3.33
	PRIMARY METAL	151.98	144.91	144.91	118.72	85.11	65.38	47.58
	SECONDARY METAL	116.83	116.48	116.48	107.49	83.26	59.30	42.83
	STEAM GENERATOR	265.94	271.52	271.52	246.31	184.16	128.13	91.96
	TOTAL CONTENTS	883.61	576.84	576.84	512.48	392.31	292.19	220.58
EFFLUENT	BREAK FLOW	0.00	313.57	313.57	392.88	577.88	771.09	1243.90
	ECCS SPILL	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	TOTAL EFFLUENT	0.00	313.57	313.57	392.88	577.88	771.09	1243.90
	*** TOTAL ACCOUNTABLE ***	883.61	890.41	890.41	905.36	970.19	1063.27	1464.48

TABLE 3.4.2-13

ENERGY BALANCE: DOUBLE-ENDED PUMP SUCTION - MAX SI

	TIME (SECONDS)	0.00	22.25	22.25	173.80	1113.90	1346.82	3600.00
				ENERGY (MILLION BTU)				
INITIAL	IN RCS/ACC/S GEN	883.61	883.61	883.61	883.61	883.61	883.61	883.61
ADDED ENERGY	PUMPED INJECTION	0.00	0.00	0.00	6.34	77.94	111.80	439.42
	DECAY HEAT	0.00	8.49	8.49	27.71	109.92	126.93	262.91
	HEAT FROM SECONDARY	0.00	-1.71	-1.71	-1.71	0.01	0.24	0.24
	TOTAL ADDED	0.00	6.78	6.78	32.34	187.87	238.98	702.58
	*** TOTAL AVAILABLE ***	883.61	890.40	890.40	915.95	1071.48	1122.59	1586.19
DISTRIBUTION	REACTOR COOLANT	302.21	15.12	16.36	35.64	35.64	35.64	35.64
	ACCUMULATOR	18.82	15.41	14.17	0.00	0.00	0.00	0.00
	CORE STORED	27.84	13.40	13.40	5.09	4.91	4.76	3.33
	PRIMARY METAL	151.98	144.91	144.91	117.96	73.86	67.28	47.69
	SECONDARY METAL	116.83	116.48	116.48	106.12	68.11	60.24	42.95
	STEAM GENERATOR	265.94	271.52	271.52	242.59	147.41	130.01	92.13
	TOTAL CONTENTS	883.61	576.84	576.84	507.39	329.93	297.92	221.74
EFFLUENT	BREAK FLOW	0.00	313.57	313.57	396.44	729.43	812.54	1352.33
	ECCS SPILL	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	TOTAL EFFLUENT	0.00	313.57	313.57	396.44	729.43	812.54	1352.33
	*** TOTAL ACCOUNTABLE ***	883.61	890.41	890.41	903.83	1059.36	1110.46	1574.06

TABLE 3.4.2-14

ENERGY BALANCE
DOUBLE-ENDED HOT LEG GUILLOTINE

	TIME (SECONDS)	0.00	25.50	25.50
		ENERGY (MILLION BTU)		INITIAL
ENERGY	IN RCS/ACC/S GEN	883.61	883.61	883.61
ADDED ENERGY	PUMPED INJECTION	0.00	0.00	0.00
	DECAY HEAT	0.00	9.53	9.53
	HEAT FROM SECONDARY	0.00	-3.95	-3.95
	TOTAL ADDED	0.00	5.57	5.57
	*** TOTAL AVAILABLE ***	883.61	889.19	889.19
DISTRIBUTION	REACTOR COOLANT	302.21	23.74	24.99
	ACCUMULATOR	18.82	11.70	10.46
	CORE STORED	27.84	11.24	11.24
	PRIMARY METAL	151.98	142.52	142.52
	SECONDARY METAL	116.83	114.17	114.17
	STEAM GENERATOR	265.94	264.66	264.66
	TOTAL CONTENTS	883.61	568.03	568.03
EFFLUENT	BREAK FLOW	0.00	321.16	321.16
	ECCS SPILL	0.00	0.00	0.00
	TOTAL EFFLUENT	0.00	321.16	321.16
	*** TOTAL ACCOUNTABLE ***	883.61	889.19	889.19

TABLE 3.4.2-15

SYSTEM PARAMETERS

PARAMETER	VALUE
Core Inlet Temperature (includes +4.85°F allowance for instrument error and deadband)	561.55°F
Initial Steam Generator Steam Pressure	968.75 psia
Assumed Maximum Containment Back Pressure	74.7 psia

3.4.2.10 References

1. WCAP-10325-A, "Westinghouse LOCA Mass and Energy Release Model For Containment Design - March 1979 Version," April 1979.

2. "Westinghouse ECCS Evaluation Model - 1981 Version," WCAP-9220-P-A, Rev. 1, February 1982 (Proprietary), WCAP-9221-A, Rev. 1.

3. EPRI 294-2, Mixing of Emergency Core Cooling Water with Steam: 1/3 Scale Test and Summary, (WCAP-8423), Final Report June 1975.

4. WCAP-8264-P-A, Rev. 1, "Topical Report Westinghouse Mass and Energy Release Data for Containment Design," August 1975.

5. ANSI/ANS-5.1-1979, American National Standard for Decay Heat Power in Light Water Reactors," August 1979.

3.4.3 <u>Containment Response Following a LOCA and MSLB</u>

CONTAINMENT INTEGRITY ANALYSES

The containment structure is designed to withstand a limited internal pressure. To ensure containment integrity, the analyses are performed to demonstrate that the containment design pressure is not exceeded should a LOCA or a secondary system pipe rupture inside containment occur during plant operation. In addition, primary or secondary pipe ruptures occurring inside a reactor containment structure may result in significant releases of high energy fluid to the containment environment, which could result in high containment temperature and pressure conditions. The high temperature and pressure can result in a failure of any equipment which is not qualified to perform its function in an adverse environment. This could degrade the effectiveness of the protection system in mitigating the consequences of the event. Thus, it is necessary to demonstrate that the conditions that can exist inside the containment during a pipe rupture do not violate the existing environmental qualification envelopes. The analysis also provides a basis for the containment integrated leak rate test (ILRT) pressure to ensure that the radiation doses will be limited to within the dose guideline values of 10 CFR Part 100 during accident conditions.

This section presents the results of the containment integrity analysis that are performed to assess the impact of power re-rate program on the containment response to a postulated LOCA or main steam line break (MSLB) accident. The containment pressure and temperature response to these postulated pipe breaks was calculated using CONTEMPT-LT/28 digital computer code [Ref 1]. Applications of the CONTEMPT-LT/28 for the calculation of the containment environmental response to a postulated primary or secondary pipe break has been accepted by the NRC staff [Ref

2]. The analyses were performed in accordance with the criteria specified in the Standard Review Plan (SRP) Section 6.2.1.1.A to meet the relevant requirements of the regulations.

3.4.3.1 Containment Integrity Analysis For a Postulated LOCA

The long term LOCA mass and energy releases that support the current licensing basis containment integrity analyses were generated with the Westinghouse 1975 M&E release model [Ref 3] and were already based upon an NSSS power of 3579 MWt. To ensure that the containment pressure and temperature responses following a postulated LOCA at rerated power conditions will be enveloped by that of the current licensing basis, the M&E releases are re-generated using the Westinghouse 1979 M&E release model [Ref 4]. This newer model incorporates improved thermal-hydraulic models which are the result of advances that occurred during the 1970's as a result of Emergency Core cooling research programs. The results of the long term M&E release analysis which are presented in Section 3.4.2 are input to the containment integrity analyses to confirm the integrity and operability of the primary containment structures and equipment necessary to mitigate the consequences of the postulated accidents.

Analyses were performed for a spectrum of possible pipe break sizes and locations at rerated conditions to assure that the worst case has been identified. Three limiting pipe breaks cases, namely; double ended pump suction guillotine (DEPS) break with minimum and maximum safety injection and double ended hot leg guillotine (DEHL) break are postulated in the analyses. The pressure and temperature profiles resulting from these three limiting pipe breaks can be seen in Figures 3.4.3-1 to 3.4.3-6. The peak calculated containment pressure and temperature are summarized in Table 3.4.3-1.

The results of the containment integrity analyses that utilize the revised M&E releases generated at rerated conditions show that the peak calculated containment pressure following a postulated LOCA is less limiting than the peak containment pressure calculated in the current WCGS licensing basis analyses and is well below the design pressure of 60 psig. As in the current licensing basis analysis, the containment pressure is reduced significantly below 50% of the peak calculated pressure for the design basis LOCA within 24 hours after the postulated accident. The LOCA containment temperature response remains bounded by the MSLB analyses and the predicted peak pressure remains conservative with respect to the specified ILRT test pressure of 48 psig.

3.4.3.2 Containment Integrity Analysis For a Postulated MSLB

The steam releases following a steamline rupture are dependent upon many possible configurations of the plant steam systems and containment design as well as the plant operating conditions and the size of the break. The variations make it difficult to reasonably determine the single absolute "worst case" for either containment pressure or temperature evaluations following a steamline rupture. This is the reason why the steam line break accident should be analyzed for a spectrum of pipe break sizes and various plant conditions from hot standby to full power.

The current licensing basis containment response analysis for a postulated MSLB considers 16 cases that were analyzed to determine the worst case containment pressures and temperatures following a MSLB.

Out of these cases, the following four cases were reanalyzed to assess the impact of plant uprating to 3579 MWt.

1. Full Double-Ended Rupture at 102% uprated power (Case 1).

2. 0.60 ft^2 Double-Ended Rupture at 102% uprated power (Case 2).

3. 0.80 ft^2 Split Rupture at 102% uprated power (Case 3).

4. Full Double-Ended Rupture at 102% uprated power, assuming failure of one MSIV (Case 16).

The method and assumptions used in the analysis for these four cases are consistent with the current design basis analysis. Limiting criteria are specified for the containment response which use the mass and energy released from the faulted steam generator as input. The LOFTRAN computer code is used to calculate the mass and energy releases following a steamline rupture. The LOFTRAN code has been reviewed and accepted by the NRC for referencing in license application for pressurized water reactors [Ref 5].

The results of the analysis indicated that the peak calculated containment pressure and temperature following a MSLB at the uprated power conditions are bounded by the current licensing basis limiting MSLB pressure/temperature cases respectively. The current pressure limiting MSLB case is a 0.80 ft^2 split rupture at 50% power (Case 9) with a peak calculated pressure of 48.9 psig and the limiting temperature case is a full double ended rupture at 50% power (Case 7) with a peak calculated temperature of 386.5°F.

Figures 3.4.3-7 to 3.4.3-14 provide the pressure and temperature profiles of the four cases reanalyzed for the power rerate program. A summary of the results is given in Table 3.4.3-2.

3.4.3.3 Conclusion

The containment response to a postulated LOCA or MSLB accident at rerated power conditions has been analyzed. The analysis results confirm that the power rerate program would not adversely impact the containment pressure and temperature response to a postulated LOCA or MSLB accident and that relevant design limits continue to be satisfied.

3.4.3.4 References

1. NUREG/CR-0255 "CONTEMPT-LT/028 - A Computer Program For Predicting Containment Pressure-Temperature Response To a Loss-Of-Coolant-Accident", March 1979.

2. NRC Safety Evaluation Related to Amendment No. 50 to Facility Operating License No. NPF-42 for the WCGS with respect to "Containment Cooling System".

3. WCAP-8264-P-A, Rev. 1, "Topical Report Westinghouse Mass and Energy Release Data for Containment Design", August 1975.

4. WCAP-10325-A, "Westinghouse LOCA Mass and Energy Release Model for Containment Design - March 1979 Version", April 1979.

5. WCAP-7907-P-A, "LOFTRAN Code Description", April 1984.

TABLE 3.4.3-1

CONTAINMENT PEAK PRESSURE AND TEMPERATURE FOLLOWING A POSTULATED LOCA AT RERATED CONDITIONS

LOCA Case	P_{max} (psig) @ t (sec)		T_{max} (^{o}F) @ t (sec)	
DEPS W/ Min SI	43.9	700	265.5	700
DEPS W/ Max SI	41.0	20	261.8	20
DE Hot Leg Guillotine	42.6	19	264.9	19
WCGS Licensing Basis	47.3	140	306.1	60

TABLE 3.4.3-2

CONTAINMENT PEAK PRESSURE AND TEMPERATURE FOLLOWING A POSTULATED MSLB AT RERATED CONDITIONS

MSLB Case	P_{max} (psig) @ t (sec)		T_{max} (^{o}F) @ t (sec)	
Full DEB	44.3	1800	383.6	107
0.6 ft2 DEB	39.9	1800	383.8	260
0.8 ft2 Split	46.0	1800	376.7	169
Full DEB/Failed MSIV	35.0	195	381.6	95
WCGS Licensing Basis	48.9	1800	386.5	125

FIGURE 3.4.3-1
CONTAINMENT PRESSURE - DEPSG LOCA W/ MINIMUM SI

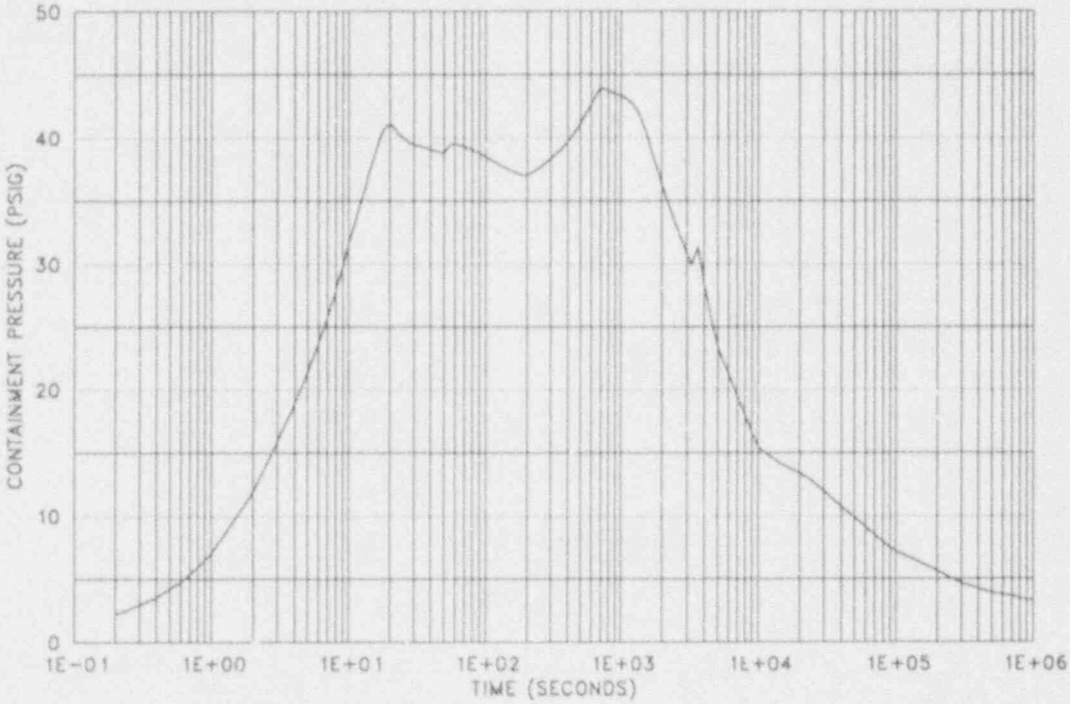


FIGURE 3.4.3-2
CONTAINMENT TEMPERATURE-DEPSG LOCA W/ MINIMUM SI

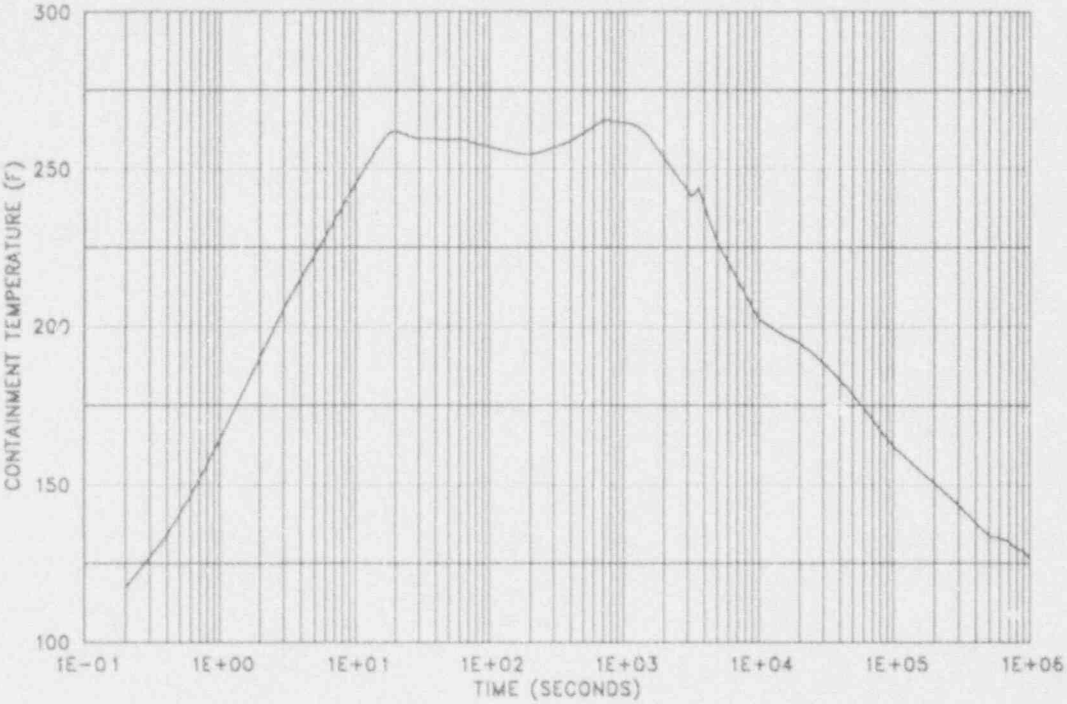


FIGURE 3.4.3-3
CONTAINMENT PRESSURE - DEPSG LOCA W/ MAXIMUM SI

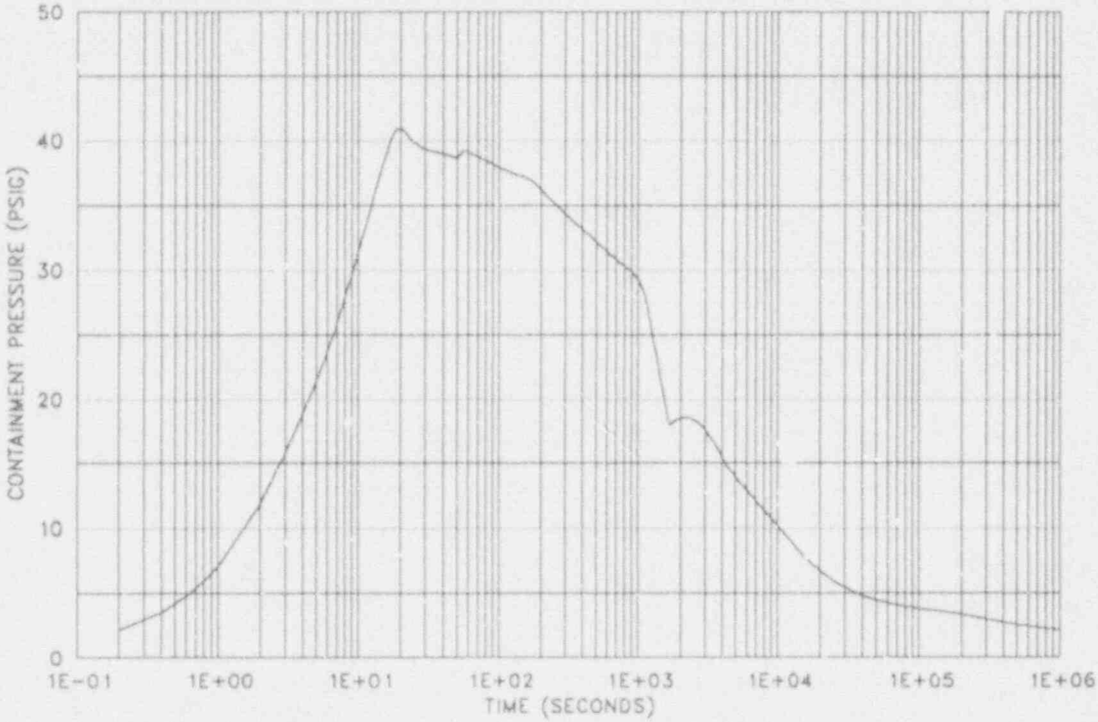


FIGURE 3.4.3-4
CONTAINMENT TEMPERATURE-DEPSG LOCA W/ MAXIMUM SI

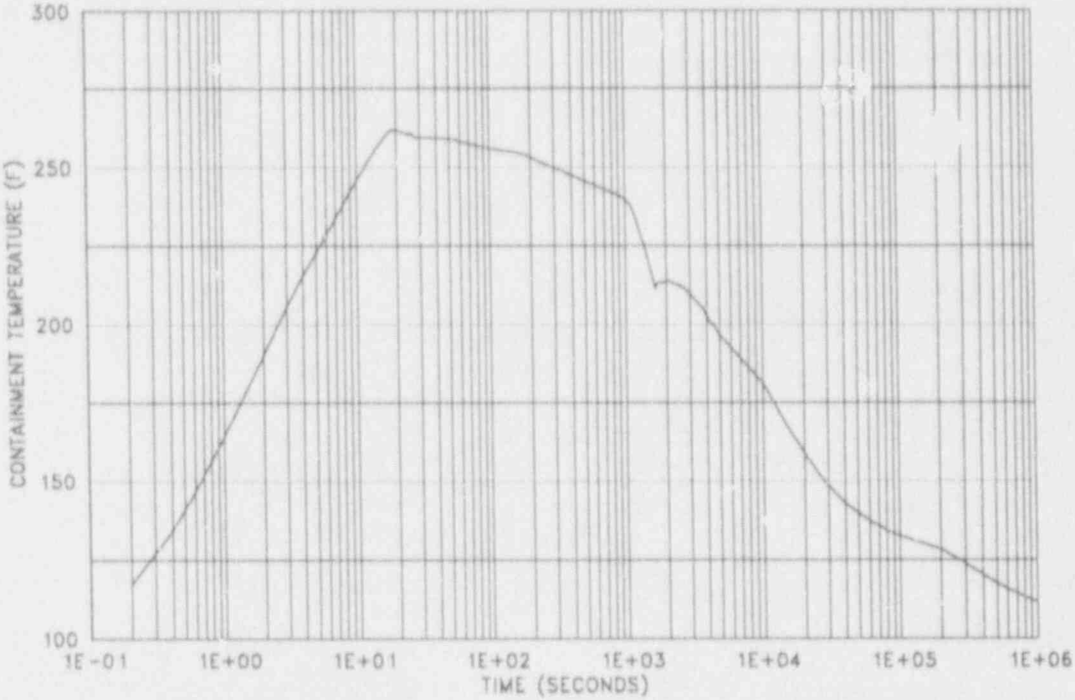


FIGURE 3.4.3-5
CONTAINMENT PRESSURE - D.E. HOT LEG GUILLOTINE

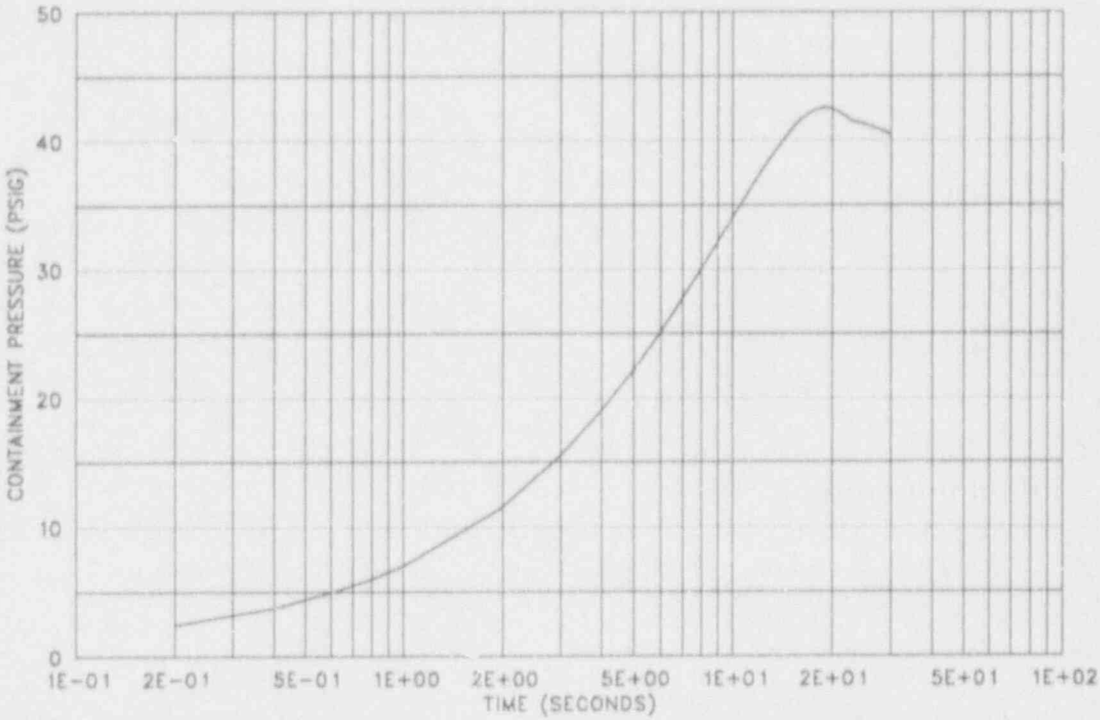


FIGURE 3.4.3-6
CONTAINMENT TEMPERATURE -D.E. HOT LEG GUILLOTINE

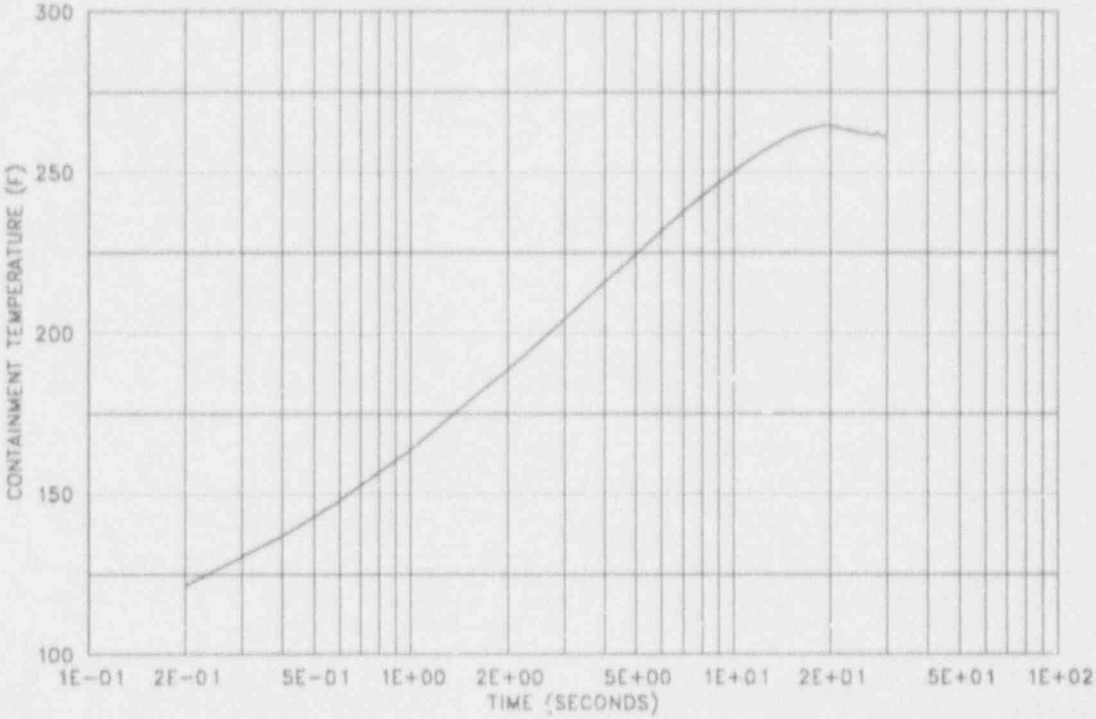


FIGURE 3.4.3-7
CONTAINMENT PRESSURE RESPONSE TO
FULL DOUBLE-ENDED MSLB AT 102% UPRATED POWER

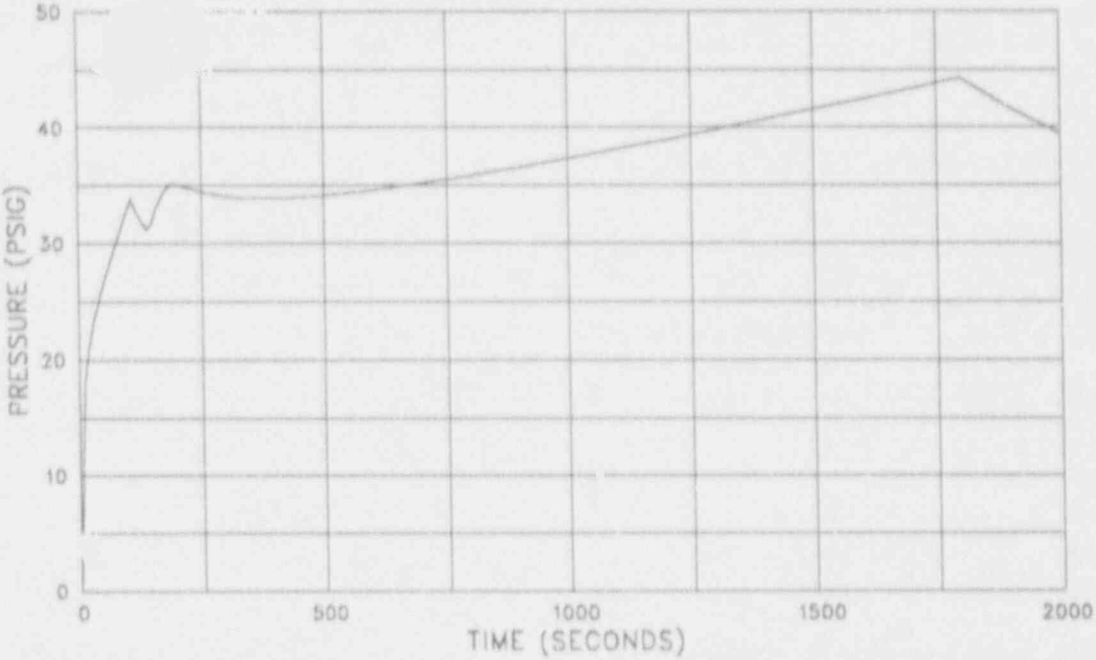


FIGURE 3.4.3-8
CONTAINMENT TEMPERATURE RESPONSE TO
FULL DOUBLE-ENDED MSLB AT 102% UPRATED POWER

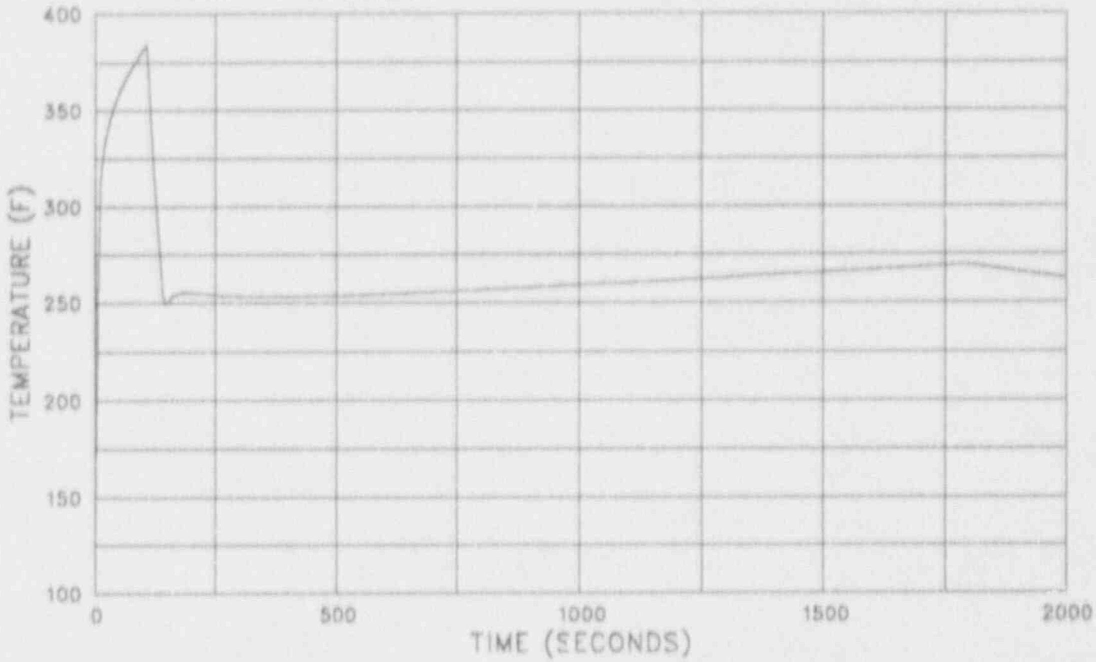
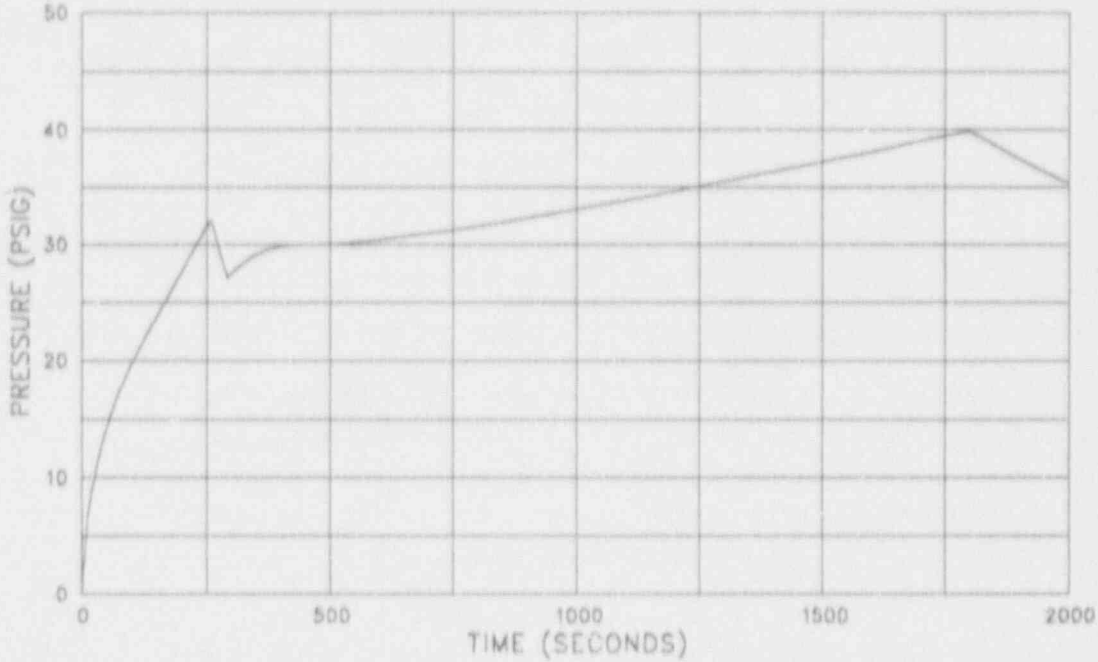


FIGURE 3.4.3-9
CONTAINMENT PRESSURE RESPONSE TO
0.6 FT2 DOUBLE-ENDED MSLB AT 102% UPRATED POWER



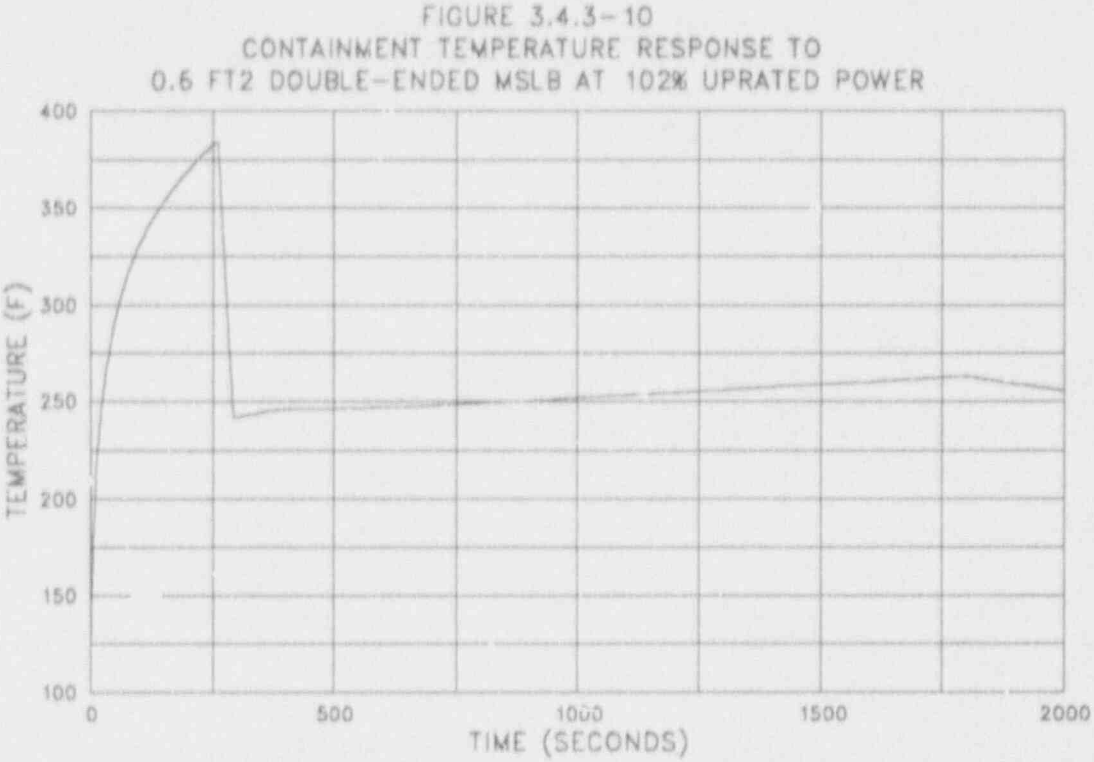


FIGURE 3.4.3-10
CONTAINMENT TEMPERATURE RESPONSE TO
0.6 FT2 DOUBLE-ENDED MSLB AT 102% UPRATED POWER
TEMPERATURE (F)
400
350
300
250
200
150
100
0
500
1000
1500
2000
TIME (SECONDS)

FIGURE 3.4-3-11
CONTAINMENT PRESSURE RESPONSE TO
0.8 FT2 SPLIT MSLB AT 102% UPRATED POWER

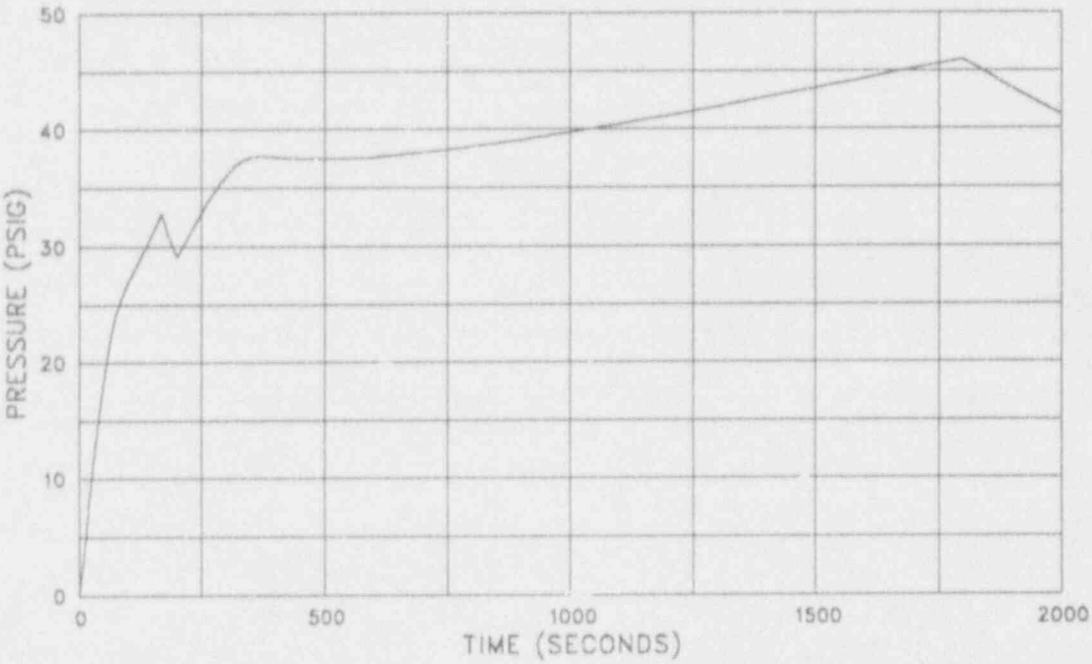


FIGURE 3.4.3-12
CONTAINMENT TEMPERATURE RESPONSE TO
0.8 FT2 SPLIT MSLB AT 102% UPRATED POWER

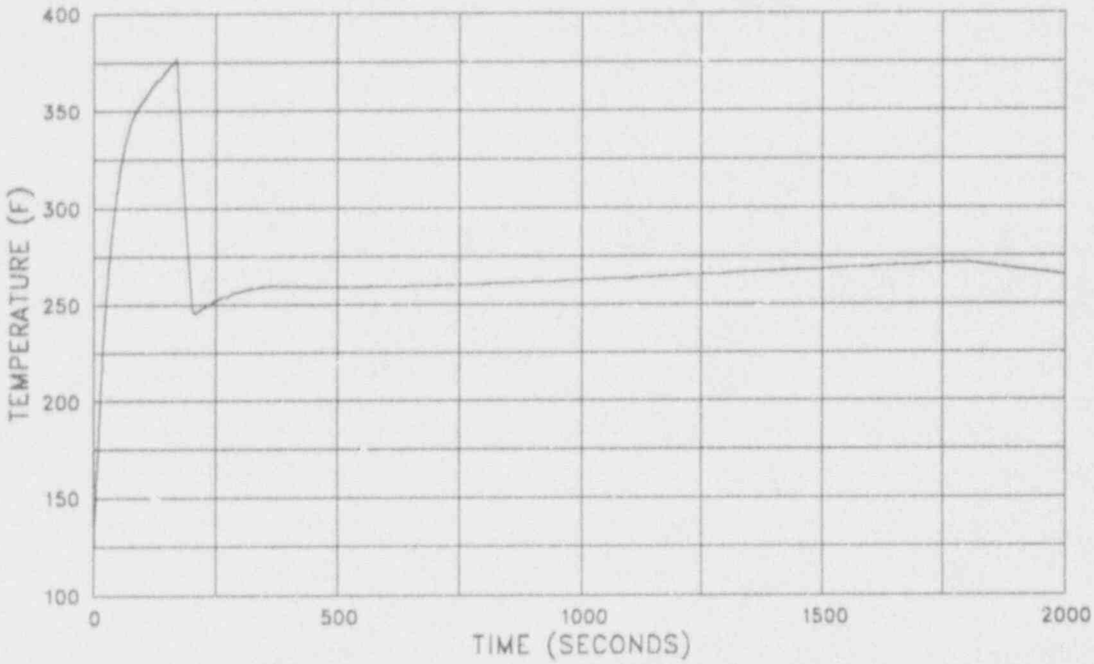


FIGURE 3.4.3-13
CONTAINMENT PRESSURE RESPONSE TO FULL DOUBLE
ENDED MSLB, FAILED MSIV AT 102% UPRATED POWER

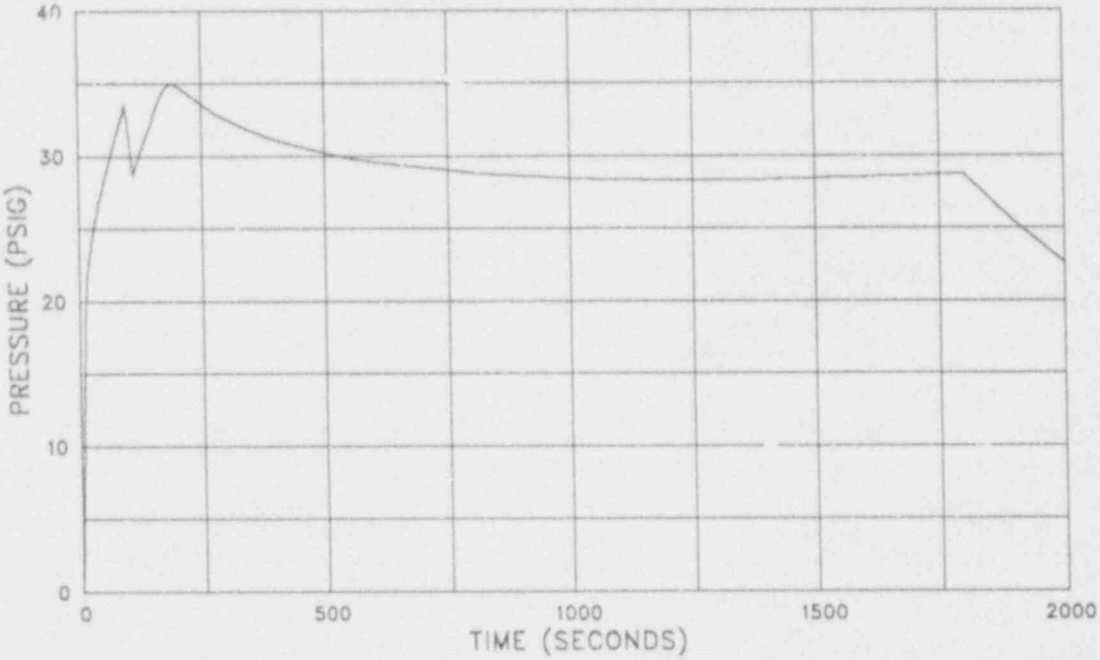
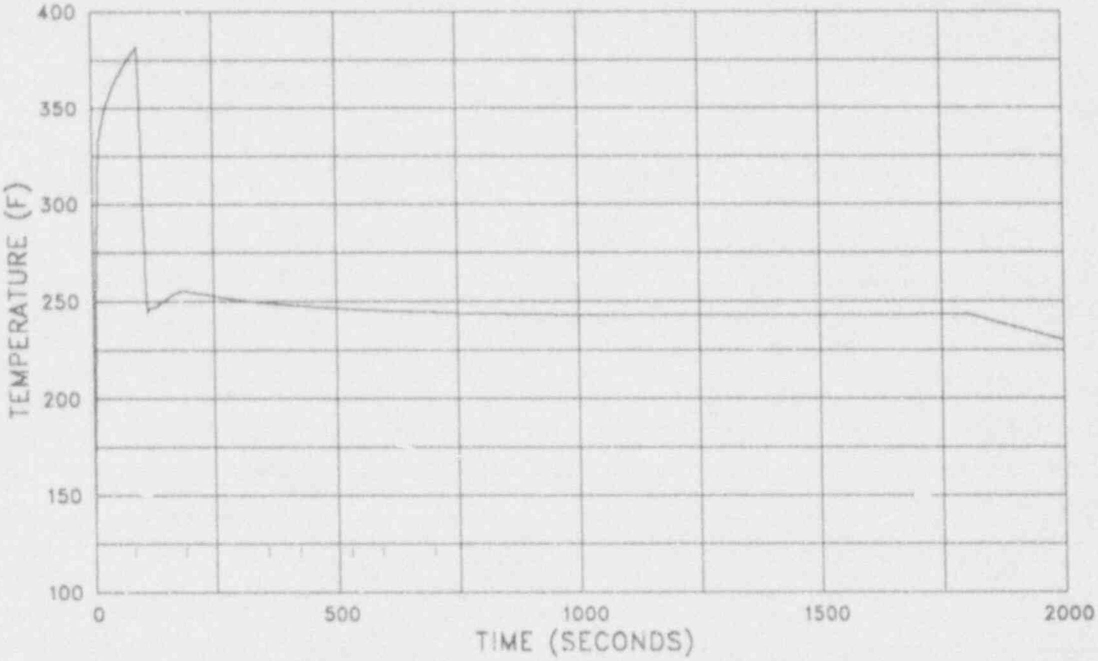


FIGURE 3.4.3-14
CONTAINMENT TEMPERATURE RESPONSE TO FULL DOUBLE ENDED MSLB, FAILED MSIV AT 102% UPRATED POWER



3.5 RCS COMPONENTS AND FLUID SYSTEMS EVALUATION

This section addresses the mechanical effects of the rerating on the primary components of the Reactor Coolant System, the Engineered Safeguards Systems, and the Auxiliary Cooling Systems.

ASME Boiler and Pressure Vessel Code, Section III, in accordance with 10 CFR Part 50, Section 55a, provides criteria for evaluation of the stress limits in components for design, normal operation, and postulated accident conditions. Any change in operating parameters has been evaluated for changes in component operating and postulated accident conditions.

3.5.1 <u>Reactor Coolant System and Connected Components</u>

3.5.1.1 Steam Generators

<u>Structural Considerations</u>

Structural evaluations of the critical components of the Model F steam generators were performed to determine the acceptability of operating at the 3579 MWth rerated conditions. The evaluations were performed according to the requirements of the ASME B&PV Code, Section III, 1971 to 1973 Editions. The critical components considered were the tubesheet and shell junction; the divider plate; the steam generator tubes; the tube/tubesheet weld; the nozzles; and the shell including the secondary manway. In the evaluations, two sets of transient parameters were considered; a high and a low T-HOT temperature. It is determined that the enveloping conditions are due to the low T-HOT temperature and a peak plugging level of 10%. These factors cause a significant reduction in the secondary side pressure, which results in a higher pressure differential across the primary to secondary boundary.

After reviewing both sets of transient parameters for the rerated conditions, the enveloping parameters (low T-HOT temperature) were used in the evaluation. The upper shell components are unaffected by variations in T-HOT and T-COLD. The primary side components, however, are affected by the variations in T-HOT, T-COLD and primary to secondary side pressure difference. The PCWG parameters (Table 2-1) and transient parameters used are assumed to be applicable for power levels of up to 3579 MWth. The evaluations consider the effects of plugging levels of up to 10%. As indicated above the applicable criteria used are from the 1971 to 1973 Editions of the ASME Code.

Discussion of Evaluations

The evaluations were based on the results of previous structural analyses of Model F steam generators. The stresses of the original analyses were scaled by the primary to secondary side pressure difference ratios (for the secondary side, steam pressure only) to determine the corresponding stresses at the rerated conditions. The temperature effects are factored into the fatigue evaluations through transient steam pressure changes.

The high T-HOT temperature conditions are close to the originally analyzed transient conditions used in the reference analyses, and therefore the resulting fatigue usages would show only slight variations from the current conditions. The reduced T-HOT temperature conditions, however, would result in increased stresses and fatigue usages. This is the result of the lower steam pressure during normal operation. For the secondary side components, although the lower pressure reduces the corresponding stresses at normal operation, it results in a higher stress range when cycling between full power and hot shutdown, because hot shutdown steam pressure is the same for both the current and rerated conditions.

Lower Shell Components

For the tubesheet evaluation, three highly stressed locations, the tubesheet center, the tubesheet and shell junction, and the tubesheet/channelhead junction were considered. The results of the evaluation showed that the maximum stress intensities calculated at these locations remain within the allowable limits for all conditions analyzed. The fatigue usage factors are also acceptable. The divider plate/tubesheet junction was analyzed for fatigue usage based on an elastic-plastic stress analysis. The fatigue usage at the junction was found to be acceptable.

For the steam generator tubes, it was determined that the maximum stresses for the critical loading conditions for the rerated case are close to those for the 100% power case. In both cases, the fatigue usage factors remain acceptable. The minimum acceptable tube wall thickness for the new pressure conditions was determined to be 0.016 inch using Regulatory Guide 1.121 ("Basis for Plugging Degraded PWR Steam Generator Tubes") guidelines. This compares to a minimum acceptable wall thickness of 0.014 inch for current conditions. The difference resulting from rerating is judged to have no effect on Technical Specification plugging criteria.

Upper Shell Components

The rerating fatigue evaluation of the upper shell considered the main and auxiliary feedwater nozzles, the secondary manway, and the steam nozzle components. These components, per the original analysis, are the limiting structural members for the upper shell region. For each component, the rerating fatigue analysis was performed utilizing the currently existing analysis of the components, scaling the stresses, and then updating the fatigue calculations if necessary. The primary stresses and maximum stress ranges were not increased by the rerating conditions because the steam pressure decreases.

In general, the impact of the rerating on the upper shell is due to the variation in steam pressure and temperature. The main feedwater temperature for all transient conditions is not significantly affected. The steam nozzle and manway bolts are the limiting parts and are expected to experience the maximum usage.

A summary of the resulting fatigue usages show that the ASME Code fatigue limit of 1.0 to be satisfied in all cases except for the manway bolts. The bolts were qualified for 20 year replacement. The rerate evaluation shows that in 15.8 years the bolts accumulate a usage of 1.0. Note that the steam nozzle appears to be slightly affected by the rerating. This is due to the conservative manner in which the load/unload cycles are considered in the analysis.

Conclusions

In conclusion, the current components of the steam generators are found to remain in compliance with the requirements of the ASME B&PV Code, Section III, for the rerated operating conditions. The secondary bolt replacement interval, however, is required to be decreased from 20 years to 15.8 years.

Additional evaluations were performed regarding the bolt replacement interval. If WCGS is operated with a 5°F T-HOT reduction (instead of 15°F T-HOT), then the recommended bolt replacement interval is 18 years.

Thermal-Hydraulic Considerations

A steam generator thermal-hydraulic evaluation of the Wolf Creek steam generators has been completed. Operating characteristics of the steam generator at the rerated conditions were calculated. Attention was focused on secondary side parameters. The calculated parameters are compared to the values for the same parameters at the current design conditions. Where appropriate, the parameter values are compared to other existing field experience. Moisture separator performance is not included since an evaluation covering the rerated conditions was provided previous to the rerating program.

The following three operating conditions were evaluated. The latter two are compared with the current design conditions:

- Current Design Conditions (100% power, no plugging)
- Uprated (104.5%) power, 10% plugging
- Uprated (104.5%) power, 15°F T-HOT reduction, 10% plugging

Operating Characteristics

Several secondary side operating characteristics can be used to assess the acceptability of steam generator operation at the rerated conditions. These parameters include circulation ratio, a measure of hydrodynamic stability, secondary mass, peak heat flux, and secondary side pressure drop.

The circulation ratio is a measure of liquid flow in the tube bundle in relation to the steam flow. It is primarily a function of power. The circulation ratio decreases by a maximum of about 10% for the reduced T-HOT condition. Since the steam flow also increases with power, the bundle liquid flow decreases by only about 8% at the same condition. The bundle liquid flow minimizes the accumulation of contaminants on the tubesheet and in the bundle. The uprating and plugging, therefore, have no material effect on this function.

The hydrodynamic stability of a steam generator is characterized by the damping factor. A negative value indicates a stable unit. That is, small perturbations of steam pressure or circulation ratio will die out rather than grow in amplitude. Damping factors are seen to remain negative at about the same level as current design. Therefore, at the rerated conditions the steam generators continue to be hydrodynamically stable.

The reduced steam pressure which results from plugging, and to a lesser degree the uprating, causes small reductions (3-7%) in steam generator mass. This results from the reduced steam pressure which in turn brings about an increased void fraction in the tube bundle. These small changes in secondary mass are not considered significant.

The value of peak heat flux shows the expected increase with power and tube plugging. For uprating, the increased total heat load is passed through the same bundle heat transfer area, thereby increasing the heat flux. For increased plugging, the same heat load is passed through a smaller heat transfer area, also increasing the heat flux. The maximum calculated heat flux is well within nucleate boiling limits and is also close to values for steam generators currently operating in the field.

The maximum increase in total secondary side pressure drop, 4 psi, for the steam generator is very small in relation to the total feed system pressure drop. This will have no significant effect on the feed system operation.

Conclusions

The thermal-hydraulic operating characteristics of the steam generators currently installed at Wolf Creek are within acceptable ranges for all rerating conditions.

U-Bend Wear Evaluation

The purpose of this evaluation is to develop estimates for the increased wear at the anti-vibration bar (AVB) intersections which could result from the implementation of the rerated conditions for Wolf Creek. The evaluation performed shows that a modest number of tubes (<10 tubes per steam generator) might be affected following long term operation at the rerated conditions.

Projection of Wear Rates

Tubes in the U-bend region of steam generator tube bundles have shown some degree of wear at the intersections with the AVBs. Rather than corrosion, AVB indications have been confirmed as wear based on examination of pulled tubes and AVB segments. The mechanism of wear is demonstrated to be fluidelastic vibration. The alternative wear mechanism, excitation by turbulence, could not account for the degree and rate of wear.

Fluidelastic stability is expressed as a ratio of the applied velocity to the critical velocity for instability. When this ratio equals or exceeds 1.0, the tube is considered fluidelastically unstable and subject to vibration.

Other than fluid conditions, the principal variable affecting fluidelastic vibration is the tube natural frequency. Tube natural frequency is primarily governed by the length of unsupported spans. Longer unsupported spans lead to lower natural frequencies and higher stability ratios. Large radius U-bends have a lower natural frequency than a small radius U-bend with identical support conditions (the same number of active AVBs). Similarly, the same tube supported by less AVBs will have longer unsupported spans yielding a lower natural frequency and higher stability ratio.

U-bends in the Model F have three anti-vibration bars. Large radius U-bends have six tube/AVB intersections and possible tube support locations. Evaluations have shown that there is a low but finite probability that one of these support locations is inactive. The probability of adjacent inactive supports is even smaller. Further, the smaller radius U-bends can be fluidelastically stable with one or more inactive AVB support locations. As a result of these factors, only a small population of tubes in the bundle is susceptible to fluidelastic vibration.

Tube wear at the AVB intersections first appears in the large radius U-bends with the highest number of unsupported intersections. These tubes have the highest stability ratios. Subsequently, wear appears in the tubes which are more stable, but still unstable with respect to fluid elastic vibration. These tubes are inner radius U-bends or those which have better support than the tubes which first display wear.

The projection curve for the incidence of tubes with AVB wear and for tubes plugged due to AVB wear is principally based on field experience for the Model 51 steam generator and analyses for various support probabilities with associated wear rates. AVB wear and plugging for these units has been tracked for more than 15 years. The plugging curve developed represents an envelope of the plant averaged plugging per steam generator. While it is possible that a single steam generator will exceed the curve, the plant average plugging experience to date has been bounded by the curve.

Model F Projected Wear

The Model F AVB plugging curve was specifically derived by considering the comparative onset of AVB wear between the Model F and the Model 51, and the relative rate of incidence of AVB wear between the two steam generator models. The Model F exhibited AVB wear earlier in operation and at a more pronounced incidence than the Model 51. The projected curve for the Model F was derived by adjusting the Model 51 projection curve based on the early Model F wear data as supported by the analysis models. With the exception of a single plant, which has been shown to be unique due to a manufacturing anomaly, all current operating experience for the Model F AVB wear is enveloped by the projection curve developed for the Model F.

Estimated Increase in Wear for Rerated Conditions:

To estimate the change in projected tube plugging which could occur as a result of a change in operating conditions, a three step approach is used.

1. Determine the change in stability of any tube given the change in operating conditions. That is, calculate a stability ratio for rerated conditions relative to current design conditions.

2. Using a probabilistic approach, estimate the proportion of tubes in any bundle which will become unstable as a result of this relative stability ratio.

3. Using field data for the distribution of wear as a function of time, calculate the increased number of tubes subject to wear.

For the U-bend region of a given tube bundle, the change in stability of a tube with respect to fluid elastic vibration occurs only due to changes in fluid flow parameters. A one dimensional technique is used to determine the stability ratio of a tube at the rerated conditions under consideration relative to the stability ratio at the current design conditions. The ratio is a function of tube natural frequency. The increased power and reduced steam pressure cause the increases in relative stability ratio.

In order to determine the number of tubes which could be affected by wear at the rerated conditions but would not have been affected at current design conditions a probabilistic approach is used. A Monte Carlo study was first used to determine the relative occurrence of a range of AVB support conditions expected in the U-bend region. Then a finite element structural analysis was used to determine the tube stability and natural frequency at current design conditions for each occurring AVB support condition. Seven different row groups were used to characterize the Model F bundle.

Coupling the latter result with the relative stability ratios permits calculation of the proportion of tubes in each row group which are stable at current operating conditions but which become unstable at either of the rerated conditions. Field data is then used to determine the percentage of total tubes wearing which is occurring in each row group at different times. This permits calculation of the total expected increase in tubes with wear which will result from changing to the rerated conditions. The calculation is performed for the several times during operation of the plant for which data are available.

The above calculation was used to define a new projected plugging curve based on the curve at current conditions. The new projection curves show an 8% and 15% increase in plugging, respectively, for uprating and uprating with T-HOT reduction. At uprated conditions with T-HOT reduction, this means that up to 8 additional tubes per steam generator could be affected by wear for long term operation.

Conclusion

The bounding condition for Rerating/T-HOT Reduction of the Wolf Creek Nuclear Plant shows a modest potential for an increase in the maximum number of tubes (about 8 tubes per steam generator) which could be affected by long term, continuous operation at the rerated conditions. The rerating, therefore, is expected to have no significant consequences on tube plugging due to wear in the U-bend region at the AVB intersections.

<u>Steam Generator Corrosion Evaluation</u>

The following is an assessment of the proposed changes on the corrosion propensity of the Wolf Creek steam generators.

Westinghouse has developed a steam generator corrosion algorithm to assist in the analysis of corrosion issues. The algorithm considers plant features, operating conditions and operating chemistry in order to provide a measure of the corrosion propensity and compute the "level of concern" for corrosion to occur. Although the algorithm can not calculate absolute corrosion rates, the relative rate change resulting from operating features, conditions or chemistry can be derived. To calculate the impact of a particular parameter on the "level of corrosion concern," a parametric evaluation can be performed by holding other parameters constant and varying only the parameter of interest. This technique has been used to evaluate the effect of sets of T-HOT and T-SAT (Saturation Temperature) pairs keeping all other input parameters constant.

Effect of Operating Parameter Changes on Corrosion

The Westinghouse steam generator corrosion algorithm provides an estimate of the relative propensity for steam generator corrosion resulting from changes in operating parameters. Beta factors for various forms of steam generator corrosion are calculated based on theory and both laboratory and field experience. Beta factor values for various forms of steam generator corrosion reflect the effect that specific plant materials of construction and specific plant operational parameters have on the propensity for these forms of corrosion. When beta factors are further combined with chemistry values, another algorithm generated parameter, Zcum, which is a level of concern for secondary side corrosion, is derived. The algorithm currently addresses PWSCC, ODSCC, denting and pitting. Before looking at specific changes in the beta factors resulting from possible operating changes for the Wolf Creek unit, a brief discussion of how changes in operating parameters can change corrosion propensity is provided.

Changes in PWR steam generator operating conditions result in changes in the absolute temperature of heat exchanger tubing, crevice chemistry concentration, and hoop stress. These factors, in many cases, control the rate of steam generator materials corrosion. The discussion below briefly examines the role of each of these factors in the corrosion mechanism.

Absolute Temperature

The corrosion rate of steam generator materials is a function of absolute temperature if an aggressive chemical environment is also present. The relationship between absolute temperature and corrosion rate is given by the Arrhenius equation,

$$\text{Rate} = Ae^{-Q/RT}$$

where A is a coefficient comprised of a number of environmental factors pertinent to the mechanism of the reaction, R is a constant equal to approximately 2 cal/mole deg., T is temperature in deg. K and Q is the activation energy in cal/mol. The impact of a temperature change on the rate of a corrosion reaction depends on the value of Q, the activation energy. For purposes of this study, Q was assumed to be 33 kcal per mole for secondary side SCC and 45 kcal per mole for PWSCC. The overall effect of absolute temperature is incorporated into the corrosion algorithm via a term that includes the Arrhenius equation.

Chemical Concentration

The concentration of a particular corrosive environment is another important factor that is incorporated into the corrosion algorithm. For secondary side corrosion issues, the concentration of a corrosive solution in a flow restricted region is a function of the "available superheat." Superheat is the difference in temperature between the primary fluid and T-SAT at a particular elevation in the tube bundle. Superheat provides the driving force for chemical concentration in crevice regions. A non-volatile contaminant species will concentrate at the liquid-vapor transition region by boiling and evaporation. The solution formed will have a boiling point elevation resulting from dissolved salts which is numerically equal to the local "available superheat." Generally speaking the greater the "available superheat," the higher is the chemical concentration in the flow restricted region and the higher is the corrosion rate. This effect is incorporated, at least partially, into the algorithm via the coefficient A of the Arrhenius equation.

As an example of the effect that the "available superheat" term can have, laboratory denting rates have been observed to double with an approximately 7°F increase in superheat.

Stress

Stress on the outside diameter of heat transfer tubing is a function of many parameters. But when T-HOT is changed and there is a resulting change in the differential pressure across the steam generator tube wall, one must consider the impact of this change on hoop stresses. Thermal stresses may also change slightly, but the changes are negligible relative to their effect on corrosion.

The influence of hoop stress changes on the rate of secondary side initiated stress corrosion cracking can be significant since stress is a component of the coefficient A in the Arrhenius equation and the dependency of SCC on stress is thought to be a function of stress to the second power. Thus, when T-HOT and T-SAT are reduced but the primary pressure remains constant there is an increase in the differential pressure across the tube wall which results in an increase in hoop stress and increased O.D. cracking propensity. The hoop stress variations that result when the differential pressure (primary-secondary) is changed can be calculated from the equation:

$$\text{hoop stress in psi} = \frac{(\text{primary P psi} - \text{secondary P psi})\ (\text{Tube ID In.})}{2\ (\text{wall thickness in.})}$$

Increased hoop stress resulting from increased differential pressure on the tube wall will also affect the PWSCC rate. However, despite the fact that PWSCC is proportional to stress to the fourth power and ODSCC is proportional to stress to the second power, the rate increase is greatest for ODSCC in the cases considered here. Since PWSCC is observed almost exclusively in highly stressed regions (e.g., expanded regions, expansion transitions, dented tube support intersections) where residual stresses (which can be as high as 40 ksi) associated with the deformations are the major contributors to total tensile stress, and since the incremental increase in hoop stress (1.4 ksi maximum) resulting from the cases in this evaluation is considered to be small compared to total stress (residual + operational > 50 ksi), the impact of hoop stress changes on PWSCC has been ignored in this evaluation.

Results

Calculations were made for the Wolf Creek steam generators using the Westinghouse corrosion algorithm for each of the cases and specific input operational parameters. The greater the value of the beta factor, the greater the propensity for the particular form of corrosion. These data, therefore, indicate that with the design features, materials combinations and operating conditions at Wolf Creek, the steam generators would be most susceptible to pitting and ODSCC and least susceptible to denting corrosion. Of course these forms of corrosion would occur only if chemistry conditions promoting them develop within steam generator crevices during operation.

To further put these results in perspective, however, a comparison of the numerical value of the respective beta-factors with those of an earlier steam generator design is required. A Model 51 steam generator (drilled carbon steel supports and mill annealed Alloy 600 tubing) which has removed copper containing alloys from the feed train and which operates at comparable conditions would have beta factors for denting and PWSCC more than two orders of magnitude higher, an ODSCC beta factor 5 to 10 times higher and a pitting beta factor approximately six to eight times as high. The low beta factor values for Wolf Creek reflect the additional corrosion protection/margin provided by materials and design features of the Model F steam generators.

To quantitatively assess the relative differences between each case evaluated in this study, the beta factor for each corrosion process was normalized by dividing by the current parameter beta factor. If the value of the normalized beta factor is greater than one, the effect of parametric changes is to increase the tendency for that form of corrosion. Conversely, a value less than one represents a decrease in the tendency. A temperature increase compared to the reference case produces a positive contribution to all forms of corrosion covered by the algorithm. However, the magnitude of the net effect will differ due to differences in activation energy for the corrosion processes and may be further modified by superheat and hoop stress considerations.

Beta factors are additive. The data shows that when referenced to the current conditions (Case 1) all cases considered in this study result in modest increases (3 to 11%) in the propensity for secondary side corrosion. Although the uprated case without T-HOT reduction (Case 3) results in a 9% increase in the propensity for all forms of corrosion, the uprated case with T-HOT reduction (Case 2) results in a 7% decrease in the propensity and the uprated case with a 5°F T-HOT reduction (Case 4) is essentially unchanged for all forms of corrosion. Although the propensity for denting corrosion in these cases is much higher, denting provides the smallest contribution to the total of all corrosion processes in this study, only 6% at maximum.

The results show that for the range of conditions evaluated in this program there is essentially no difference in the propensity for pitting. The marginally lower beta factors for Cases 2 and 4 are principally due to lower temperatures and suggest only a slight temperature dependence of pitting.

As already stated, increases in temperature and hoop stress will increase the rate of ODSCC. In comparing the operating parameters, it is clear that the increased temperature and differential pressure across the tube wall in Case 3 (Uprate without T-HOT reduction) results in a 12% increase in the relative rate of ODSCC when compared to Case 1 (current parameters). Case 2 (Uprate with 15° T-HOT reduction) compared to the current operating condition results in a relative rate for ODSCC of 95% of the Case 1 value. In Case 2 the net effect of the two opposing factors, decreasing temperature and increasing tube wall differential pressure, results in a net reduction in ODSCC corrosion propensity. The increase in hoop stress offsets some of the benefit that would be achieved by the temperature reduction alone. In Case 4, temperature again decreases and differential pressure increases. But in this instance, the differential pressure effect is greater than the temperature effect and results in a 2% increase in the propensity for ODSCC and a 4% increase in propensity for all forms of secondary corrosion. A higher corrosion propensity alone does not mean that ODSCC will occur in the Wolf Creek steam generators. The results simply mean that if an appropriate crevice chemistry is produced during operation, the rate at which ODSCC will progress will be greater for that case.

The analysis for PWSCC shows that uprating without T-HOT reduction (Case 3) results in a 4% increase in the propensity for PWSCC and that uprating with concurrent T-HOT reduction can result in a 35% (Case 2) or 22% (Case 4) reduction in PWSCC propensity relative to current operating conditions. The much greater benefit of T-HOT reduction for PWSCC versus ODSCC is due to the higher activation energy (stronger dependence on temperature change) and the lack of a strong dependence on hoop stress for PWSCC.

For denting, the controlling factor in determining the relative corrosion rates is available crevice superheat (difference between primary and secondary temperatures). An analysis of algorithm output shows that, in all cases, using the current conditions as a reference point, a large increase in the propensity for denting is predicted. This interpretation is questionable in that the superheat dependence of carbon steel corrosion has been extended to 405 stainless steel. In fact, in all Westinghouse laboratory testing of 405SS, both isothermal and with heat transfer, no evidence of accelerated growth of magnetite has been observed, even in environments known to be aggressive denting environments for carbon steel. (Accelerated growth of magnetite, where the volume of magnetite produced is greater than the volume of the base metal consumed, is required to produce denting). Corrosion of the stainless steel can occur in these aggressive environments but denting has not resulted. This is not to say that denting can not occur in steam generators with 405SS supports. Based on our current knowledge, however, we know of no crevice chemical environment which is expected to produce it.

The anticipated benefits or detriments due to operating Wolf Creek at the conditions discussed previously do not reflect the full impact of secondary chemistry on the individual corrosion process. The benefits achieved by operating at a reduced T-HOT value may be rapidly eliminated if a chemical environment conducive to acceleration of any of the secondary corrosion processes forms in superheated crevices. Likewise, operation with any set of conditions which increases the propensity for secondary side corrosion requires establishment and maintenance of an appropriate crevice chemistry before corrosion will occur.

Zcum is an algorithm-generated parameter which is derived from beta factors, the knowledge of how secondary chemistry affects various corrosion processes, and specific plant chemistry conditions. It provides a cumulative "level of concern" (Zcum) for secondary side corrosion as a function of time. To be explicit for Wolf Creek, actual Wolf Creek secondary chemistry parameters would have to be entered. Since the effort required to provide that plant specificity is clearly beyond the scope of this program, some level of chemistry control must be assumed. Two cases are considered here: (1) operation at the EPRI secondary chemistry guideline limits, and (2) operation at 25% of the EPRI limits.

Even though terms like "level of concern" are nonquantifiable with respect to measurable steam generator materials corrosion they provide a mechanism for making decisions relating to cost-benefit on changing operating conditions. The cost of changing operating conditions may be weighed against delaying potential corrosion related repairs, increasing availability and extending years of service.

This may be illustrated by using the plots in these two figures to determine the time required to reach the same "level of concern" for all forms of secondary side corrosion considered in the algorithm. The results clearly show the benefit that chemistry improvements and maintenance of chemistry at ALARA values can have on slowing secondary side corrosion. However, since the effects of chemistry are cumulative, once corrosion has initiated, chemistry improvements cannot reverse or lower the "level of concern." Similarly short duration excursions where chemistry parameters are elevated can become significant contributing factors to achieving levels of concern for secondary corrosion much earlier than would be expected based on "normal or average" chemistry control levels.

Conclusions

1. The propensity for corrosion of the Wolf Creek steam generators, as represented by beta factor values generated in this evaluation, are much lower than corresponding beta factor values for earlier steam generator designs operated at comparable conditions. A Model 51 steam generator comparably operated would have an ODSCC beta factor 5 to 10 times larger, a pitting beta factor 6 to 8 times as high, and denting and PWSCC beta factors more than two orders of magnitude higher. The lower corrosion propensities for Wolf Creek steam generators are due to use of more corrosion resistant materials and incorporation of corrosion limiting features in their design and fabrication.

2. Because steam generator corrosion phenomena are thermally activated processes, the higher T-HOT value associated with uprating without T-HOT reduction (Case 3) increases the relative corrosion propensity for all forms of steam generator corrosion except pitting. Operation at conditions covered by this case results in a 9 and 11% increase in secondary side and total corrosion propensity. Individual corrosion propensity increases of 4% for PWSCC, 12% for ODSCC and a large increase for denting results from these operational changes. Although the increase in the calculated relative propensity for denting is large, the contribution of denting to the total corrosion propensity is relatively small, amounting to less than 6%.

3. For the two cases where uprating is combined with a reduction of T-HOT, the results, compared to the current operating condition, are a 7% decrease in corrosion propensity for all forms of corrosion and a 3% increase in secondary side corrosion (Case 2), and a 4% increase in secondary corrosion propensity with no change in total corrosion propensity (Case 4). For each form of corrosion individually, pitting propensity remains essentially unchanged, PWSCC is lower by 22 to 35%, denting is higher, and ODSCC is lower by up to 6% in one case and 2% higher in the other.

4. Simply because secondary side corrosion propensities increase or decrease does not mean that secondary corrosion will or will not occur. Regardless of the calculated relative corrosion propensity before any corrosion will occur a susceptible material must be subjected to an environment which will promote corrosion. Many of the benefits associated with operational changes to decrease corrosion propensity can rapidly be negated by changes in chemical

and physical components of the steam generator crevice environment resulting from adverse changes in chemistry control. For that reason, an ALARA philosophy of chemistry control must be supported and rigorously implemented to minimize the potential for steam generator corrosion.

3.5.1.2 Reactor Vessel

Evaluations were performed for the various regions of the Wolf Creek reactor vessel to determine the stress and fatigue usage effects of the NSSS operation at the rerated conditions throughout the current plant operating license. The evaluations assess the effects of the rerating design transients, operating parameters, and reactor vessel/reactor internals interface loads on the most limiting locations with regards to ranges of stress intensity and fatigue usage factors in each of the regions identified in the reactor vessel analytical reports and addenda. The evaluations assume that the plant may continue to operate at the high or low temperature rerated conditions or in accordance with the original design basis or somewhere in between for the entire licensed life. Where appropriate, revised maximum ranges of stress intensity and maximum usage factors were calculated for the rerating. In other cases the original design basis remains conservative so that no new calculations were necessary, and the design basis maximum ranges of stress intensity and fatigue usage factors continue to govern.

The rerating had only a minimal effect on the maximum ranges of stress intensity and maximum fatigue usage factors. In fact, all of the maximum ranges of stress intensity for the original design basis transients and operating parameters were unchanged as a result of the rerating. Moreover, only two of the maximum usage factors increased due to the rerating. In all other cases, the usage factors were either unchanged or conservative when the revised designed transients were considered. Overall, the maximum effect of the rerating on the reactor vessel analytical report is an increase in the maximum cumulative fatigue usage factor for the core support lugs of only 0.001. The only other increase in usage factor is 0.0007 for the outlet nozzle support pads.

The faulted condition seismic and LOCA loads at the reactor vessel/reactor internals interface were evaluated by comparison with the applicable generic interface load set. All of the loads were found to be lesser in magnitude than their corresponding generic load components. Therefore, no stress calculations were required in order to incorporate the revised interface loads.

Based upon the satisfactory results of the rerating evaluations as discussed above, the Wolf Creek reactor vessel is acceptable for plant operation in accordance with the Wolf Creek power rerate program. Considering any combination of the design basis and the rerating RCS transients for the specified numbers of occurrences, the reactor vessel and fatigue analyses and evaluations justify operation with a range of vessel outlet temperatures from 603.2°F up to 620.0°F and a range of vessel inlet temperatures from

538.2°F up to 558.8°F. Such operation may continue for the remainder of the plant license.

An evaluation of the impact of rerating on reactor vessel integrity was performed for Wolf Creek. Neutron fluence changes and other relevant system parameters associated with rerating were considered in the evaluation. Review of the applicability of the current heatup and cooldown curves previously generated by Westinghouse indicate that these curves were generated in accordance with the irradiation embrittlement prediction of Regulatory Guide 1.99, Revision 2 (Draft). Based on a comparison of RT_{NDT} values of limiting belt line region materials at 1/4T and 3/4T locations of the vessel, it is concluded that the applicability dates of heatup and cooldown curves will not be impacted due to the implementation of the rerating program.

The Wolf Creek reactor vessel beltline region material properties were confirmed against the latest available information. The properties defined from the latest information are consistent with those used in prior Wolf Creek submittals to the NRC relative to complying with the requirements of the PTS Rule.

Revised PTS analyses were performed for the rerating using the latest procedure specified by the NRC in the PTS Rule and the calculated neutron fluence values for the rerated conditions for Wolf Creek.

Material chemistry (Cu, Ni content) and unirradiated properties for the Wolf Creek reactor vessel indicates an end of life (32 Effective Full Power Years) RT_{PTS} of less than 140°F which is well below the PTS screening criteria (270°F for plates, forgings and longitudinal welds; 300°F for circumferential welds). Therefore, the rerating program for Wolf Creek has no significant adverse impact on the RT_{PTS} evaluation.

For Wolf Creek, the RT_{PTS} values of the limiting plate materials are well below the Emergency Response Guidelines criteria. Hence, there will not be any revisions to the Emergency Response Guidelines due to the rerating program.

Surveillance capsule withdrawal schedules have been evaluated for the uprated condition. It has been determined that there will not be any change in the surveillance capsule withdrawal schedule for the Wolf Creek.

3.5.1.3 Reactor Vessel Internals

An evaluation and analysis was performed to investigate the impact of NSSS power rerating on Wolf Creek reactor internals. In order to assess the impact of the plant operating modifications, the following specific analyses were performed:

-RPV Thermal-Hydraulic Analysis
-RPV LOCA Blowdown Analysis
-Reactor Internals Flow Induced Vibration Assessment
-Reactor Internals Seismic Evaluation
-Control Rod Drop Time/Rod Insertability Evaluation
-Component Thermal Loads and Stress Evaluation
-Component Structural and Fatigue Evaluation

These analyses indicated that the structural adequacy of the Wolf Creek reactor internals is not adversely impacted by the implementation of the aforementioned program.

3.5.1.4 Control Rod Drive Mechanism

The original design report establishes the structural integrity of the pressure boundary components of the Model L-106A1 Control Rod Drive Mechanism (CRDM) including the seismic sleeve and the capped latch housing assembly for the Wolf Creek Generating Station as required by Section III of the ASME Boiler and Pressure Vessel Code, 1974 Edition through Winter 1974 Addenda. To determine the effect of the rerating the revised primary side parameters and the associated NSSS design transients were compared to those analyzed in the original design report. The evaluation results indicate that the rerating temperature ranges have a diminishing and insignificant effect on the thermal analysis and that the revised design transients are bounded by those analyzed in the original design report.

Therefore, the Wolf Creek Model L-106A1 CRDM remains in compliance with the Westinghouse and industry codes and standards which were originally applicable when Wolf Creek was initially licensed.

3.5.1.5 Reactor Coolant Pumps

The original design report established the structural integrity of the pressure boundary components of the Model 93A-1 Reactor Coolant Pump for the Wolf Creek Generating Station. The results contained in the design report determined that all of the specific requirements in the generic design specification and project specific design specification were satisfied. The impact of the rerating parameters and revised design transients on the original design report was evaluated.
At the rerated conditions the steam generator outlet (pump inlet) temperatures are 556.6°F (maximum T-HOT) and 538.0°F (15°F T-HOT) as compared to 558.6°F used previously. The specified operating (zero load) temperatures remain unchanged at 557°F. A maximum 20.6°F (558.6°F to 538.0°F) T-HOT reduction has a diminishing and insignificant effect on the thermal analysis performed on the RCP design.

A review of the rerated transients confirmed that they remain bounded by the analyzed transients of the Wolf Creek specific design report and hence the original analysis remains applicable.

Therefore, the pressure boundary components of the Wolf Creek Model 93A-1 Reactor Coolant Pump continue to comply with the Westinghouse and industry codes and standards which were originally applicable when Wolf Creek was initially licensed.

Based on the rerating parameters, revised loads for the Wolf Creek RCP Motors were determined for each of the three design criteria cited below. The revised loads were then incorporated into the analysis that confirmed the ability of the RCP Motors for:

1. Continuous operation at the new hot loop load (7092 HP). The new hot loop load point exceeds the nameplate rating by 92 HP. This should produce an increase in stator temperature of less than 3°C which remains within acceptable limits.

2. Continuous operation at the new cold loop load (9074 HP). This represents a 324 HP increase over the nameplate rating of the motor. It is expected that this load will cause less than a 5°C increase in stator winding temperature during operation which is considered acceptable.

3. Rotor winding temperatures during the worst case starting scenario. The worst case starting scenario is a cold loop, 80% voltage start with 30,100 gpm reverse flow.

    a. The calculated rotor winding temperature rises (based on a conservative all heat stored analysis) are within the design allowances and are therefore acceptable.

Based on compliance with the applicable design criteria the motors are suitable for operation at the revised loads associated with the rerating.

3.5.1.6 Pressurizer

The Wolf Creek pressurizer equipment specification and stress report have been evaluated relative to the rerating parameters and the revised NSSS design transients. The evaluation shows that the rerating thermal parameters and transients conditions are enveloped by the generic transients used in the analysis of the pressurizer components. Therefore, the stress analysis results presented in the Wolf Creek pressurizer stress report are still applicable for all the components.

Thermal Evaluation

The review of the rerating parameters and revised NSSS design transients shows that the Hot Leg and Cold Leg fluid temperatures are revised for some of the transients. The pressurizer parameters such as fluid/steam space temperatures, pressures, insurges and outsurges and sprays are not revised.

The changes in the Hot Leg water temperature affect the pressurizer components in the lower head region and lower portion of the shell during insurges. The changes in the Cold Leg water temperature affect the spray nozzle and the upper head/shell area where sprays impinge during some of the spray nozzle operations. These changes affect stresses in the pressurizer components which are different than those calculated in the stress report for the loadings provided in the Design Specification.

Stress Analysis and Fatigue Evaluation

The pressurizer components stress analysis and fatigue evaluation involve the determination of the component thermal and mechanical service loads for the rerating/T-HOT reduction conditions. These loads are compared to the loads used in the analysis of the Wolf Creek Pressurizer stress report. If the thermal and mechanical loads of the rerating/T-HOT conditions are not enveloped by the loads to which the pressurizer components were analyzed originally, then the stresses are to be recalculated and revised fatigue usage factors are determined.

A comparison of the Wolf Creek rerating/T-HOT reduction thermal/mechanical loads and the original analysis loads showed the following:

- Pressurizer fluid and steam temperatures were not revised for the rerating/T-HOT reduction program.

- Component mechanical loads such as nozzle pipe loads, support skirt loads, valve support bracket loads and seismic lug loads are not revised for the rerating/T-HOT reduction program.

- Hot Leg and Cold Leg fluid temperatures were revised for the rerating/T-HOT reduction program. The changes in the Hot Leg fluid temperatures affect the pressurizer fluid/Hot Leg fluid differential temperature during insurges and thus affect thermal stresses in the lower head components. The changes in the Cold Leg fluid temperatures affect the pressurizer steam/Cold Leg fluid differential temperature during sprays and thus affect thermal stresses in the spray nozzle and upper head/upper shell area where the sprays impinge upon.

However, a review of the Hot Leg and Cold Leg fluid temperatures for the rerating/T-HOT reduction program showed that the resultant differential temperatures between

Hot Leg/Cold Leg fluids and pressurizer fluids are enveloped by the differential temperatures to which the pressurizer components were analyzed. Therefore, new thermal stresses and fatigue usages were not calculated. Also, new fracture mechanics analysis was not required to be performed.

Summary

The results show that the Wolf Creek Pressurizer components meet the ASME Code, Section III stress analysis requirements for the rerating/T-HOT reduction program loadings.

3.5.1.7 Piping and Supports

An evaluation has been completed for the Wolf Creek RCS loop piping and supports to determine the magnitude of the rerating on the qualification of the following components: the reactor coolant loop piping, the primary equipment supports, the primary equipment nozzles, the Class 1 auxiliary line piping, and the Class 1 loop branch nozzles.

The rerating parameters indicate temperatures in the loop different than originally analyzed. These differences have a possible impact on the thermal, seismic, and LOCA analyses. The normal 100% power operating temperature changes for this program will result in a slight change to the thermal displacements and loads in the loop. The changes in thermal loadings in the loop piping are negligible. The impact of the change in thermal displacements on the boundary conditions for the Class 1 auxiliary line piping is also negligible.

The gaps in the primary equipment supports will change by a very small amount due to system thermal contraction caused by the lower temperatures. The tolerance on measuring the gaps in the primary equipment supports is of approximately the same magnitude as the change associated with the reduced T-HOT operating conditions.

Since the gap condition of the primary equipment support system is assumed to change by a negligible amount, the seismic loadings on the loop piping and primary equipment support system do not change by a significant amount. The assumption is that the system configuration, within an acceptable tolerance, is the same as that used in the original design basis analysis.

Two of the inputs into the loop LOCA analysis are the hydraulic forcing functions (HFF) and the reactor pressure vessel (RPV) LOCA displacements. The original design basis for the primary loop piping and support evaluation included loadings that correspond to large doubled-ended guillotine breaks. Wolf Creek has implemented loop leak-before-break (LBB) which eliminated the requirement to postulate the large guillotine breaks. There is, therefore, significant margin available to accommodate the nominal increase in the hydraulic forcing functions and the RPV LOCA displacements. The existing loop LOCA analysis for Wolf Creek is sufficiently bounding so that the loads and displacements that are output from the analysis apply for both the upper and lower bound T-HOT operating conditions. As with the seismic condition, the primary equipment support gaps do not change enough to significantly impact the LOCA results.

As part of the piping reconciliation, the fatigue analysis was reviewed to determine if the rerating parameters would impact the results. The input of interest is the set of thermal design transients for the system. These thermal transient changes are factored into the fatigue evaluation to determine that none of the allowables are exceeded. Although some of the stress ranges increase by incorporating these changes, the overall fatigue results remain in compliance with all applicable industry and ASME code requirements.

The pressurizer surge line was reviewed relative to thermal stratification. The surge line is welded into the loop hot leg and is therefore exposed to the reduced hot leg temperature. The magnitude of the stratification is proportional to the difference between the pressurizer temperature and the hot leg temperature. This temperature range is increased for a T-HOT reduction program and has been considered relative to the analysis performed for this condition. The results of that review indicate that the new temperature ranges in the surge line have negligible impact on the existing analysis.

The loop loadings used in the LBB evaluation include combinations that contain both thermal and seismic loads. There is negligible change in the thermal and no change in the seismic loads. There are no changes in the loop loadings used in the loop LBB evaluation for the rerating program.

An additional consideration was evaluated for the rerating program, namely the influence of the interferences noted in the reactor coolant pump tie rods and cross-over leg whip restraints. The analysis and evaluation performed as a result of the "noise" resolution program have been factored into the rerating program. In summary, the loop support interferences in the noise program had the opposite influence on the support gaps as the T-HOT reduction. Because the T-HOT reduction tends to shrink the system by a small amount, the effect is to relieve loads that were generated by the interferences present in the system. The rerating program does not yield a loading condition more severe than already documented in the noise program or the original design basis. The fatigue usage factors, which are tied to the design thermal transients, accounted for both the noise program loadings and the modified design thermal transient conditions of the rerating program.

For the components assessed there is negligible impact from the rerating/T-HOT reduction conditions on the existing analysis results. These components are acceptable for the operation at the rerated conditions without any modifications to the systems involved.

3.5.1.8 Auxiliary Cooling System

Auxiliary Heat Exchangers and Tanks

An evaluation has been performed to determine the effects of the Wolf Creek performance capability parameters and design transients on the qualification of NSSS auxiliary heat exchangers and tanks. The only tanks for which design transients have been identified are the boron injection tank (BIT) and the safety injection accumulators. Due to BIT elimination at Wolf Creek the boron injection tank is not being used for its original design function and hence is not impacted by the rerating. Therefore, only the safety injection accumulators have been reviewed.

The original design parameters and qualification requirements for the Wolf Creek auxiliary heat exchangers and safety injection accumulators are defined by the equipment specifications and purchase order documents. The auxiliary heat exchanger specification and purchase order documents did not require the letdown chiller, seal water, moderating, and reactor coolant drain tank heat exchangers to be qualified for pressure or temperature transients. The transients were not included in the design because they were expected to have no effects on these components. Therefore, this equipment was designed only for maximum steady state pressures and temperatures.

The regenerative heat exchangers, letdown heat exchanger, letdown reheat heat exchanger, excess letdown heat exchanger, residual heat exchangers, and safety injection accumulator tanks qualification requirements were evaluated at the rerated conditions and were found still bounded by the original Wolf Creek design parameters.

Therefore, there are no new limitations associated with the operation of the NSSS auxiliary heat exchangers and tanks at the Wolf Creek rerated conditions.

Auxiliary Valves

The qualifications of the auxiliary valves were evaluated to determine the impact of the Wolf Creek rerating. The original design criteria and parameters for the Wolf Creek auxiliary valves were defined by SSDC 1.3X with the valve specifications listing the applicable transients. The auxiliary valves were evaluated at the rerated conditions and were found still bounded by the original Wolf Creek design parameters.

Therefore, compliance with all applicable requirements is continued and there are no new limitations associated with the operation of the NSSS auxiliary valves at the Wolf Creek rerated conditions.

Auxiliary Pumps

The original design parameters and qualification requirements for the Wolf Creek auxiliary pumps are defined by the equipment specifications and purchase order documents as noted in SSDC 1.3X with the pump specifications listing the applicable transients. The auxiliary pumps were evaluated at the rerated conditions and were found still bounded by the original Wolf Creek design parameters.

Therefore, compliance with all applicable requirements is continued and there are no new limitations associated with the operation of the NSSS auxiliary pumps at the Wolf Creek rerated conditions.

3.5.2 Fluid Systems

The fluid systems evaluations are based on NSSS operation at a power rating of 3579 MWth and other parameters presented in Table 2-1. The acceptability of systems and affected components are discussed system by system.

The review of the Wolf Creek Reactor Coolant System and Auxiliary Fluid Systems show that operation at the proposed rerated conditions is acceptable. The Component Cooling Water System heat loads and cooling water functional requirements of BOP-FR-1 bound the rerated conditions.

3.5.2.1 Reactor Coolant System (RCS)

The reactor coolant system process flow diagram and the RCS process parameter tables describe the fluid conditions of the NSSS during plant operation. The rerated RCS fluid conditions formed the basis for later system and equipment evaluations. Since the RTD Bypass was eliminated prior to the rerating, these lines and flows are removed from the RCS process flow diagram. The RCS loop fluid conditions of pressure, temperature and flow were calculated for the increased power for both the normal T-HOT and the 15°F T-HOT reduction operating conditions. It should be noted that the RCS normal T-HOT rather than reduced T-HOT fluid conditions are limiting for system and component evaluations. Both the normal T-HOT and the 15°F T-HOT parameters for the uprated power conditions are tabulated to provide for plant operation between these temperature and flow conditions.

3.5.2.2 Pressurizer

The capacities of the pressurizer safety valves, the pressurizer power operated relief valves, the spray valves, and pressurizer heaters, required for the rerated conditions remain at or below the rated capacities of the existing installed equipment. Therefore, no changes are required for any of this equipment. The pressurizer spray line flow requirement of 900 gpm was verified for the available driving pressure head and the calculated spray line flow resistance. The pressurizer surge line pressure drop proof of design calculation using the rerated RCS fluid conditions and L/D resistance criteria were used to show that the pressure at the bottom of the reactor vessel will not exceed 110% of the design pressure. This calculation assumed that each of the three pressurizer safety valves were relieving at 103% of set pressure and discharging at rated capacity of 420,000 lb/hr. The pressure drop from the pressurizer safety relief valves to the reactor coolant system were evaluated. It has been determined that the pressure and pressure drops are acceptable.

3.5.2.3 Reactor Coolant Pumps

The RCP motors are rated at 7000 hp. The highest power consumption occurs when the RCS is at cold conditions with high water density which is not effected by the uprating. At hot conditions power operation conditions, the RCP motor horsepower is less. The change in fluid density at the pump operating point, at 100% power for the uprated conditions, is so small that the power consumption will be nearly identical to existing requirements.

3.5.2.4 Chemical and Volume Control System (CVCS)

The CVCS interfaces with the RCS include the regenerative and excess letdown heat exchangers. The design inlet temperature for each heat exchanger is 560°F from the RCS. Because the rerated conditions are below the design inlet temperatures, the performance of both heat exchangers are acceptable. The regenerative and excess letdown heat exchanger performance was calculated for the rerated conditions to document that the calculated temperatures are within the outlet design temperature criteria.

3.5.2.5 Residual Heat Removal System (RHRS) and Safety Injection System (SIS)

Operation at the uprated reactor power level of 3565 MW_{th} will increase the decay heat load on the residual heat removal system. The effect of the increased decay heat load on plant cooldown was evaluated. The ANSI/ANS-5.1-1979 standard has been used for the decay heat generation. This decay heat standard has lower heat generation for the first 24 hours after shutdown than the decay heat curve used in the original RHRS design analyses. For normal cooldown using both residual heat removal loops, calculations show the reactor coolant system can be cooled from 350°F at 4 hours after shutdown to 140°F within 17 hours after shutdown, which is within the design basis of 20 hours. This two train cooldown time is only slightly longer than the previous 3411 MW_{th} reactor power rating 16 hour cooldown time.

10 CFR 50 Appendix R (Fire Protection) requires plant cooldown to 200°F cold shutdown within 72 hours with minimal equipment available. This is interpreted as single RHRS train failure. For single train RHRS cooldown, calculations show the reactor coolant system can be cooled from 350°F at 4 hours after shutdown (4 hours was assumed to conservatively reflect decay heat levels when the RCS reached 350 ^{o}F) to 200°F within 26 hours after shutdown. This single train cooldown compares with a cooldown time of 12 hours of the previous analysis provided in the USAR Section 5.4.7.

The safety injection system (SIS) including the centrifugal charging pumps, safety injection pumps, associated piping valves and related equipment has been analyzed to support operation at the uprated power and T_{HOT} reduction conditions. Based on all reviews, the power uprating and T_{HOT} reduction will not affect the ability of the system to perform its intended safety function.

3.5.2.6 Other Fluid Systems

The Boron Recycle System (BRS) functions to collect borated radioactive effluent from the RCS and to process the effluent into reusable water and boric acid. Any RCS boron concentration changes for the uprating are not significant and do not impact the design of the BRS.

The Boron Thermal Regeneration System (BTRS) is designed for a daily load cycle which is only a reference cycle. Any RCS boron concentration changes for the uprating are not significant and do not impact the design of the BTRS.

Gaseous Waste Processing System (GWPS) and the Liquid Waste Processing System (LWPS) design and operation will not be effected by the rerating. The Gaseous Waste Processing System and the Liquid Waste Processing System design bases included radioactive source terms based on 3565 MW_{th} reactor power rating. These source terms are still bounding for the systems designs.

<u>System Descriptions</u>

A review of the Westinghouse fluid system descriptions (RCS, CVCS, RHR, SIS, BTRS, GWPS and LWPS) shows the system descriptions need not be revised for the uprating. The RHRS design basis and components do not change for the uprated power. For the Reactor Coolant System Description the only potential change is in the system parameter section. This change is not considered necessary for the Westinghouse system description.

Auxiliary Equipment Design Transients

The Wolf Creek rerating parameters have been reviewed concerning the NSSS Auxiliary Equipment Design Transients SSDC 1.3X, Rev. 0 and the Auxiliary Heat Exchangers Specifications ASME Sections III and VIII. The conditions and assumptions in the design transients and the heat exchanger specification are unchanged or still bounding. The new operating temperatures are encompassed by the original design transients.

Emergency Power Requirements

The emergency power requirements for the diesel generators remain unchanged for Westinghouse supplied equipment with respect to the time and manner with which these electrical loads are sequenced on the diesels. The only electrical equipment potentially affected by the uprating are the pressurizer heaters. The requirements for the pressurizer heaters are based upon a desired heatup rate for the pressurizer of approximately 50°F/hr. Hence, there are no changes in the pressurizer heater requirements.

Heat Loads and Cooling Water Requirements

Westinghouse issued the Component Cooling Water System (CCWS) Functional Requirements and Design Criteria Standard for CCWS BOP-FR-1, Rev. 3. This standard contained the design criteria and guidelines used by the Architect Engineer (Bechtel) for the design of the CCWS.

In reviewing the components cooled by the water system it is possible to identify a number of components which are not affected by the rerating in terms of their cooling flows and heat loads, which therefore require no further consideration. For the Westinghouse supplied items these include: RHR pump seal coolers, Centrifugal charging pump bearing oil coolers, Safety injection pump bearing oil coolers, Reactor Coolant Pumps, Reactor coolant drain tank heat exchanger, Seal water heat exchanger, Recycle evaporator package, Positive displacement charging pump oil cooler, Waste gas compressors, Catalytic hydrogen recombiner, and Waste evaporator package. This leaves the RHR heat removal heat exchangers, Excess letdown heat exchanger and the Letdown heat exchanger that are affected by the rerating. The CCWS flows and heat loads were calculated for the letdown and excess letdown heat exchangers for the rerated T-COLD RCS conditions for CCWS normal operations and 4 hour shutdown operations. The residual heat removal heat exchanger CCWS heat loads were calculated for 4 hours after shutdown and during post LOCA recirculation. The changes in the CCWS flows and heat loads are not significant.

Precautions Limitations & Setpoints (PLS)

The Wolf Creek Rerating Program required the review of the PLS (Precautions, Limitations and Setpoints) document. This section documents the fluid systems review and changes to the PLS to reflect the operation of the Wolf Creek Plant at the rerated conditions. The Wolf Creek PLS document was reviewed for impact for the NSSS uprating to 3579 MWth on the systems and components. The review verified that the only PLS impact was on the RCS pressurizer spray line connection (Low Spray Line Temperature Alarm) and that there is no impact on any other auxiliary system process setpoints.

3.5.2.7 Main Steam System

The rerating will not affect the system design pressure (1200 psia). The steam mass flow rates and the steam velocities at full power will increase for both Cases in Table 2-1, but these increases should have no significant effect on the system components, piping, and piping supports, except in the case of the MSIV closure event which is discussed in the Balance of Plant section.

Steam Generator Safety Relief Valves

Since the system design pressure does not change because of the rerating, there is no need to revise the lift setpoints of these valves. The combined relief capacity of these valves must be such that the valves will relieve the maximum amount of steam generated in the limiting case loss of heat sink event (usually a turbine trip with a delayed reactor trip) without allowing the pressure in the main steam system to exceed 110% of the 1200 psia design pressure. Since the steam generation rate in the limiting case loss of heat sink event is seldom known, as is the case with the Wolf Creek upratings, Westinghouse recommends, conservatively, that the relieving capacity of these valves be equivalent to 105% of the engineered safeguards steam flow for the plant. It has been customary in evaluating past plant reratings to consider that the full power steam flowrate for the rerated condition is the engineering safeguards steam flowrate. For the proposed Wolf Creek rerating, the steam flowrate of 15.92×10^6 lb/hr for Case 1 represents the worst case. Thus, the combined relieving capacity of the steam generator safety relief valves must be at least 105% x 15.92×10^6 lbs/hr = 16.74×10^6 lbs/hr. Table 10.3-2 of the Wolf Creek USAR shows the combined relief capacity of the steam generator safety relief valves to be 18.23×10^6 lbs/hr, which is more than adequate for the rerated conditions.

Steam Generator Power Operated Relief Valves (PORVs)

The adjustable setpoint of the PORVs is set between the zero load steam generator pressure (1107 psia) and the setpoint of the lowest set steam generator safety relief valve (1200 psia). The relief capacity of the PORVs must satisfy two criteria: first, they must be capable of relieving 10% of the full power steam flow without causing the lowest set safety relief valve to lift (required to satisfy the 50% load rejection capability of the plant), and second, they must have sufficient relief capacity to accommodate a 50°F/hr plant cooldown rate. The engineering evaluation has determined that the PORVs should easily be able to accommodate the small increase in steam that must be discharged during a plant cooldown in the uprated conditions.

Steam Dump Valves (or Turbine Bypass Valves)

To meet the 50% load rejection criteria, these valves must be capable of discharging 40% of the full power steam flow again without causing the main steam system pressure to increase to the setpoint of the lowest set safety relief valves. The engineering evaluation has determined that the steam dump valves will be capable of relieving the additional amount of steam (5%) required by the uprating at a valve inlet pressure below the setpoint of the lowest set steam generator safety relief valve.

3.5.2.8 Condensate and Feedwater System

The increased feedwater flow rates and the increased final feed water temperatures associated with the rerated conditions will cause a general increase in the fluid velocities throughout these systems. However the fluid velocity increases will be small and will not adversely affect the piping or components in the Condensate and Feedwater systems.

Feedwater Isolation Valves (FIVs)

The 5 second rapid closure requirement for these valves will be retained for the rerated conditions. The ΔP across the FIV valve discs and the thrust on the valves, the system piping and the piping supports that occur during a rapid closure, are proportional to the fluid density and the square of the fluid velocity through the valve at the time of closure. It is estimated that these forces will increase by approximately 5% over those that would occur under current operating conditions, for both the Case 1 and Case 2 reratings. The engineering evaluation has determined that the small increases in the forces can be accommodated.

Feedwater Control Valves (FCVs)

The increased mass flow rate, increased volumetric flow rate and increased final feedwater temperatures associated with the uprated conditions will not adversely affect these valves.

In assessing the capability of the FCVs to accommodate the rerated feedwater flow conditions, it was assumed that the feedwater pump speed control program presently used for the current plant operating conditions would be retained for the rerated conditions. The feedwater pump speed control program will compensate for the decrease in steam generator pressures associated with the rerated conditions, and the result will be that the pump pressure head available to force the feedwater through the system piping from the feedwater pump discharge to the steam generator feedwater inlet will be the same as for the current operating conditions. The feedwater volumetric flowrate (gpm) increases by 5.1% for the Case 2 rerating, and by 5.7% for the Case 1 rerating. Noting that the flow losses in the feedwater piping, exclusive of the flow loss in the FCV, are proportional to the square of the volumetric flow rate, it was determined that the piping flow losses would increase by 10.5% for Case 2 and by 11.7% for Case 1. This means of course that the FCVs must compensate for the increased flow losses in the feedwater system piping by moving to a more open position than the position they hold at full power for the current plant operating conditions. From past experience with plant reratings of a similar magnitude, wherein the feedwater flow rate increases a similar amount (5 to 6%), it has been found that the FCVs have sufficient capacity to accommodate the increased feedwater flows and associated increased feedwater piping flow losses. On this basis it is judged that the FCVs in the Wolf Creek Generating Station will be able to accommodate these increased flows and increased feedwater piping flow losses. It should be noted that if there is a problem with the FCV capacity to handle the full power feedwater flow rates in the rerated conditions, the problem can be solved by modifying the feedwater pump speed control program so as to provide more pump discharge pressure at the full power rerated conditions.

A more severe requirement for the FCVs is that when in the full open position, they must be capable of passing 103% of the full power feedwater flow rate with the steam generator pressure 75 psi above the full load steam pressure. For the Case 2 rerating, this means that the individual FCV must have sufficient capacity to allow a feedwater flow of 103% x 15.83×10^6 lbs/hr / 4 = 4.076×10^6 lbs/hr at a steam generator pressure of 807 psia + 75 psi = 882 psia. For the Case 1 rerating each FCV must be capable of passing 103% x 15.92×10^6 lbs/hr / 4 = 4.10×10^6 lbs/hr at a steam generator pressure of 950 psia + 75 psi = 1025 psia. Satisfying these criteria is necessary to prevent the plant from tripping following a 50% load rejection due to low-low steam generator water levels. It should be noted that when the 50% load rejection transient occurs, the feedwater pumps rapidly speed up to provide increased feedwater pump discharge pressure to help satisfy these criteria, and the FCVs have moved to their full open position. It has been found however, that in other plant reratings of similar magnitude, that these criteria have always been satisfied with the existing FCVs. On this basis, it is judged that the FCVs in the Wolf Creek Generating Station will have sufficient capacity to satisfy the 50% load rejection criteria in the rerated conditions.

The FCVs also have a rapid (5 second) closure requirement that will be retained for the rerated conditions. As with the FIVs, the increased volumetric feedwater flowrates in the uprated conditions will cause the delta P across the valve discs and the thrust loads on the valves, piping, and piping supports to increase slightly over these forces that would be experienced during a rapid closure of the FCVs under current operating conditions. These forces will increase by approximately 5% in both the rerated conditions.

Feedwater Pump Speed Control Program

This program controls the feedwater pump speed (rpm) and consequently the head - flow curve of the feedwater pumps throughout the operating range. As can be seen in the discussion of the FCVs above, there are feedwater system effects to be considered whether the existing pump speed control program is retained or a revised program is recommended.

3.5.2.9 Auxiliary Feedwater System

Minimum Auxiliary Feedwater System Flow Rates

The minimum auxiliary feedwater flow rates that must be delivered to the steam generators for accident mitigation are based on core power, and the directly related decay heat generation rate, and other general plant parameters such as plant metal and water volumes and steam generator water volumes assumed to exist at the time of the accident. While the general plant parameters will not change, there will be a 4.5% increase in the core power level (from 3411 MWth to 3565 MWth) for both rerated conditions, with a proportional increase in the decay heat generation rate. It was found that there was sufficient margin in the specified minimum auxiliary feedwater flowrates to accommodate the increased core power and the increased decay heat generation rate associated with the rerated conditions.

Auxiliary Feedwater Pump Delivery Capabilities

The auxiliary feedwater pumps must be capable of delivering the minimum required auxiliary feedwater flows to the steam generators within one minute of actuation with the steam generators at a pressure equivalent to the set pressure of the lowest set safety relief valves plus 3% accumulation pressure. Since there is no need to change the minimum required auxiliary feedwater flow rates, and there is no need to change the setpoints of the safety relief valves, the auxiliary feedwater pump delivery capabilities will be adequate for the rerated conditions.

Auxiliary Feedwater Storage Requirements

A minimum volume of water must be maintained in the condensate storage tank for exclusive use by the auxiliary feedwater system. This volume is proportional to the core decay heat that must be removed (directly related to the core power) and the sensible heat that must be removed to accomplish a plant cooldown from zero load hot standby conditions to the conditions at which the Residual Heat Removal System (RHR) can be actuated. Using WCGS plant specific nomographs and the uprated core power level (3565 MWt), it was conservatively determined that a minimum of 240,000 gallons of water would be required to hold the Wolf Creek plant in a condition of Hot Standby for 4 hours with steam discharge to the atmosphere concurrent with total loss of offsite power, and then accomplish a 50°F/hr plant cooldown from zero load Hot Standby conditions to the conditions necessary for RHR initiation. This volume does not include allowances for water not usable because of tank discharge line isolation or other physical characteristics or uncertainties. Section 3.7.1.3 of the WCGS Technical Specifics (page 3/4 7-6) requires that 281,000 gallons of water be available in the condensate storage tank. Therefore, an adequate water reserve should be available for plant operation in the rerated conditions.

3.5.2.10 Steam Generator Blowdown and Sampling System

The blowdown and sampling system will not be adversely affected by the proposed rerating. Blowdown and sampling flow rates are dependent on the steaming rate and upon the chemistry conditions in the secondary side of the plant. Since the steaming rates increase in the rerated conditions, the blowdown and sampling rates can be expected to increase slightly. Further, for both rerated cases, but particularly for the Case 1, the pressure at the inlet of the blowdown pipe inside the steam generators will be less than that under current operating conditions. This combination of increased blowdown flow and decreased blowdown system inlet pressure will probably require an adjustment of throttle valves in the blowdown and sampling system. It is judged that there is adequate adjustment margin in the system throttle valves to accommodate the rerated conditions.

3.6 BALANCE OF PLANT

I INTRODUCTION

(Note: For clarification, the term "uprating" in this report refers specifically to an increase in reactor power level, whereas the term "rerating" refers to the power level increase in addition to the sum total of all other parameter changes (e.g., T_{Hot} reduction) to be incorporated in the power rerate program.)

The Wolf Creek Generating Station (WCGS) is currently licensed to operate at a reactor power level of 3411 Megawatts-thermal (MWt). Wolf Creek Nuclear Operating Corporation (WCNOC) has proposed uprating the plant to operate at a reactor power level of 3565 MWt, which is a 4.5% increase in reactor power, and a corresponding increase in NSSS power from 3425 MWt to 3579 MWt. The power rerate program goal is to implement a plant power uprate of 4.5%, in combination with a Reactor Coolant System (RCS) Hot Leg temperature (T_{Hot}) reduction of 5^{o}F.

Power rerating involves changes in plant heat rate and process flow rates, pressures and temperatures in the power generation cycle. Power rerating also affects thermal, hydraulic and electrical loads on various auxiliary systems and equipment. In support of the WCGS Power Rerate Program, evaluations were completed on the capability of the WCGS Balance-of-Plant (BOP) systems listed in Table II-1, components and design features to support plant operation at the 104.5% reactor power uprate conditions, in conjunction with T_{Hot} reductions of 0^{o}F, 5^{o}F and 15^{o}F. Evaluation of power rerate with no reduction in T_{Hot} (Case 1NB in Table III-1), and with a 15^{o}F reduction in T_{Hot} (Case 3NB in Table III-1) provide process parameters and evaluation results bounding the power rerate program target operating condition of 104.5% reactor power and a 5^{o}F reduction in T_{Hot} (Case 2NB in Table III-1).

In order to determine the impact of power rerating on the WCGS BOP and support systems, a systems review was conducted. The Valves Wide Open (VWO) turbine-cycle heat balance, and the power rerate parameters (see Table III-1), were used as the basis for the performance review of plant systems, components and design features.

In addition to the review of systems, a review of selected programs was performed to evaluate the impact of changes due to power rerate. The following specific issues were included in the scope on programs review:

A. The impact of power rerate on the pressurizer sub-compartment pressure and temperature consequences of a Pressurizer Surge Line Break

B. The impact of power rerate on radiological source terms used for Environmental Qualification and Radiological Shielding design

C. The impact of power rerate on the WCGS ALARA Program

D. The impact of power rerate on Hazards, including piping dynamic loads, pipe break and flooding

Section II of this report describes the scope of the Systems Review; the basic criteria used to complete the Systems and Programs Reviews is discussed in Section III. The results of the Systems Review are provided in Section IV, and the results of the Programs Review are provided in Section V. Section VI of this report provides the overall conclusions of the effort.

II SCOPE OF SYSTEMS REVIEW

BOP systems review for the WCGS Power Rerate Program included the following general areas:

Turbine - Cycle Heat Balance

Secondary Plant and Supporting Systems Reviews

WCGS Programs Review

A complete list of the plant systems within the scope of the Systems Review is provided as Table II-1. WCGS systems not listed in Table II-1 were addressed as part of the NSSS scope, or are considered to be unaffected by Power Rerating.

Table II-1

WOLF CREEK GENERATING STATION
POWER RERATE SYSTEMS REVIEW LIST

System	Class[1]	Description	System	Class[1]	Description
AB	Q	Main Steam	GN	Q	Containment Cooling
AC	Q	Main Turbine	GR	N	Containment Atmosphere Control
AD	N	Condensate	GS	Q	Containment Hydrogen Control
AE	Q	Main Feedwater	GT	Q	Containment Purge
AF	N	Feedwater Heater Extraction, Drains, & Vents	HA	S	Gaseous Radwaste
AK	N	Condensate Demineralizer	HB	Q	Liquid Radwaste
AL	Q	Auxiliary Feedwater			
AN	S	Demineralized Water Storage & Transfer	HC	S	Solid Radwaste
AP	Q	Condensate Storage & Transfer	HE	S	Boron Recycle
BL	Q	Reactor Makeup Water	HF	S	Secondary Liquid Waste
BM	Q	Steam Generator Blowdown	MA	N	Main Generator
BN	Q	Borated Refueling Water Storage	MB	N	Excitation & Voltage Regulation
CA	N	Steam Seals	MR	N	Startup Transformer
CB	N	Main Turbine Lube Oil	NB	Q	4.16 KV AC Lower Medium Voltage (Class 1E)
CC	N	Generator Hydrogen and Carbon Dioxide	NE	Q	Standby Diesel Generator (Class 1E)
CD	N	Generator Seal Oil	NG	Q	480 V AC Low Voltage (Class 1E)
CE	N	Stator Cooling Water	NK	Q	125 V DC Power (Class 1E)
CF	N	Lube Oil Storage, Transfer & Purification	NN	Q	120V Instrument AC Power (Class 1E)
CG	N	Condenser Air Removal (see System AD)	PA	Q	13.8 KV AC Higher Medium Voltage
CH	N	Main Turbine Control Oil	PB	N	4.16 KV AC Lower Medium Volt.
DA	N	Circulating Water	PG	N	480 V AC Low Voltage (non-Class 1E)
EA	N	Service Water	PJ	N	250 V DC (non-Class 1E)
EB	N	Closed Cooling Water	PK	N	125 V DC (non-Class 1E)
EF	Q	Essential Service Water including Ultimate Heat Sink	PN	Q	Instrument AC Power
EG	Q	Component Cooling Water	RJ	N	Balance of Plant Computer
EN	Q	Containment Spray	RM	N	Process Liquid Sampling and Analysis
FC	Q	Main Feedwater Pump Turbines (see System AE)	SJ	Q	Nuclear Sampling
GB	N	Central Chilled Water			
GG	Q	Fuel Building HVAC			
GL	Q	Auxiliary Building HVAC			

1. Q = System is safety related or has safety related components.
   S = System is special scope or has special scope components, but has no safety related function or components.
   N = System is non-Q, non-special scope, and has no safety related or special scope components.

III CRITERIA

Power Rerate Parameters

The Systems Review was based on the applicable process system conditions listed in Table III-1. System, equipment and component performance were evaluated at the most limiting of the proposed power rerate case(s) parameters.

Table III-1

WOLF CREEK GENERATING STATION POWER RERATE PROGRAM PARAMETERS

PARAMETER	LICENSED (100%) REACTOR POWER	VWO	(UPRATED) 104.5% REACTOR POWER CASE 1 NB 0F T_{Hot} Reduction	CASE 2 NB 5F T_{Hot} Reduction	CASE 3 NB 15F T_{Hot} Reduction
NSSS Power (MWt)	3425	3562	3579	3579	3579
Reactor Power (MWt)	3411	3548	3565	3565	3565
RCS Pressure (psia)	2250	2250	2250	2250	2250
Steam Generator					
Steam Temperature (^{o}F)	544.6	541.5	540.8	533.1	519.4
Steam Pressure (psia)	1000	975	970	908.7	807
Steam Flow, Total (10^6 #/hr)	15.14	15.85	15.94	15.89	15.83
Feedwater Temperature(^{o}F)	440	444.5	446	446	446
Zero Load Temperature (^{o}F)	557	557	557	557	557
SG Tube Plugged (%)	0	0	0	0	10

The Turbine-Generator output, and other pertinent assumptions which apply to each of these cases, are summarized in Table IV.A-2.

Uprated Power Level

An uprated NSSS power level of 3579 MWt, which is a 4.5% increase from the current licensed NSSS power level of 3425 MWt, is to be implemented based on the conditions specified in Table III-1 above. This NSSS power level corresponds to the licensed reactor power level of 3411 MWt (rated thermal power) plus 14 MWt of non-nuclear heat input from Reactor Coolant Pumps.

Evaluation Criteria

The current WCGS design criteria/bases were utilized to investigate the impact of the power rerate program parameters on plant systems and components. These design criteria and bases are provided in the industry codes and standards, NSSS and BOP design criteria and regulatory commitments which form the design bases of the Wolf Creek Generating Station, as documented in the WCGS USAR, Technical Specifications, design drawings, system descriptions, design calculations, equipment procurement specifications and related supplier documents, and current plant configuration.

IV PLANT SYSTEMS REVIEW

A. Main Power Cycle and Auxiliary Systems

1. Main Steam (AB)

The function of the main steam system is to supply steam, generated in the steam generators, to the turbine-generator system and auxiliary systems for power generation. The system provides an assured source of steam to operate the turbine driven auxiliary feedwater pump during emergency conditions. The following main steam system components were reviewed to determine their capabilities to support plant operation at the proposed power rerate conditions:

Table IV.A-1

MAIN STEAM SYSTEM COMPONENT DESIGN PARAMETERS

Components	Original Design Parameters
Main Steam Safety Valves (MSSVs)	796,500 #/hr/valve
Main Steam Isolation Valves (MSIVs)	3,964,231 #/hr @ 1.05 psid
Power Operated Relief Valves (PORVs)	594,642 #/hr/valve
Turbine Bypass Valves (TBVs)	528,571 #/hr/valve
Main Steam Flow Transmitters	-52 thru 343 inch wc
Main Steam Piping	15,136,752 #/hr, 100% Flow 15,850,801 #/hr/, VWO

Main Steam Safety Valves (MSSVs)

Each of the four main steam lines is provided with five spring-loaded safety valves. The first safety valve set pressure is 1185 psig (1200 psia), which corresponds to the steam generator design pressure, per ASME Code requirements. The remaining safety valves are set at higher pressures so all safety valves are open at full relief capacity without exceeding 110% of the steam generator design pressure.

The MSSVs are required to pass 105% of the Engineered Safeguards Design (ESD) steam flow at a pressure not to exceed 110% of the steam generator design pressure. Westinghouse recommends the MSSVs be capable of relieving 105% of the maximum power rerate main steam flow, which is 15,940,000 #/hr for Case 1NB, or, 1.05 x 15,940,000#/hr. = 16,737,000 #/hr. The combined relieving capacity of the existing safety valves is equivalent to 18,229,608 #/hr, and thus, sufficient for power rerate. Based on the preceding, the existing Main Steam Safety Valves are adequate to support power rerate.

Main Steam Isolation Valves (MSIV)

To isolate the non-safety related portions of the main steam system and to assure that steam is available to operate the turbine-driven auxiliary feedwater pump for reactor cooldown following a loss of main feedwater, a main steam isolation valve is provided in each of the four main steam lines outside of the containment, downstream of the MSSVs. The valves were originally designed to pass 105% (VWO) steam flow and close against the steam generator no-load pressure of 1107 psia.

The MSIV flow capacity and capability to close were reviewed at the most limiting power uprate condition, Case 1NB. Due to the increase in steam flow rates and the decrease in main steam pressure, the steam velocity through the MSIVs will increase at power rerate conditions. However, the increase in steam velocity will not adversely affect the MSIVs' operation. The MSIVs were originally designed based on an inlet steam pressure equal to the steam generator no-load pressure of 1107 psia, and a mass flow rate approximately four times the VWO flow in the forward direction, and approximately ten times the VWO flow in the reverse direction. Since the increase in steam flow due to power rerating is small (~0.57% (15.85×10^6 #/hr @ VWO vs. 15.94×10^6 #/hr @ Case 1NB)) and the steam line pressure decreases, power rerate will not adversely impact MSIV closure times, or the capability of the valves to close and remain closed. Therefore, it can be concluded that the MSIVs are adequate to support power rerate.

Power Operated Relief Valves (PORVs)

One power operated relief valve is provided on each of the four main steam lines to control steam generator pressure during startup, load changes and shutdown, when the main steam isolation valves are closed or when the turbine bypass system is not available. The valves were originally sized to relieve 15% of VWO steam flow (15.85×10^6 #/hr) at steam generator no-load pressure (1107 psia), and to pass sufficient flow at all steam generator pressures to achieve a 50^oF per hour plant cooldown.

Westinghouse recommends that the minimum combined relieving capacity of the PORVs be 10% of the plant design steam flow rate. This represents a maximum steam flow capacity of 398,500 #/hr per steam line at the maximum power rerate condition (Case 1NB). Since the existing PORVs are sized to relieve 15% of the VWO main steam flow, or 594,642 #/hr per steam line, the existing PORVs are adequately sized for the proposed power rerate conditions.

The set pressure for the PORVs shall be between the no-load steam generator pressure, which is 1107 psia for the power rerate, and the set pressure of the lowest set main steam safety valve which is 1185 psig. The existing PORV set pressure of 1125 psig falls within these values, and therefore, the existing set pressure is adequate for power rerate.

Turbine Bypass Valves (TBV)

The turbine bypass valves are provided to enable the NSSS to follow turbine load reductions by dumping steam to the HP, IP, and LP condensers. The twelve turbine bypass valves were originally sized to pass a total of 40% of VWO steam flow, and thus, permit the turbine to take a 50% load rejection without a reactor trip. These valves also allow a turbine and reactor trip from full load without lifting the main steam safety valves.

Each turbine bypass valve was originally designed with a flow capacity of 528,571 #/hr. at 970 psia, and a maximum capacity of 970,000 #/hr at 1200 psia. In order to satisfy their functional design basis, the turbine bypass valves must be capable of relieving 40% of the power rerate full load main steam flow, at a pressure below the setpoint of the lowest set MSSV.

Based on a review of the turbine bypass valve relieving capacity at each of the power rerate conditions, the valves are adequately sized to satisfy this design basis. Therefore, the turbine bypass valves are adequate to support power rerate.

Main Steam Flow Transmitters (FT-512 through FT-543)

Two main steam flow transmitters are provided on each main steam line to transmit input signals to the feedwater control system to provide for feedwater pump speed control and feedwater control valve throttling. Based on a comparison of the instrument ranges to the most limiting power rerate process conditions (Case 1NB), the existing instruments have sufficient range to operate properly at the proposed power rerate conditions.

Main Steam Piping

The main steam piping from the steam generators to the high-pressure turbine and second-stage reheaters is designed for 1200 psia and 600°F and was sized based on the VWO flow rate of 15.85×10^6 #/hr. The main steam piping is seismic Category I up to and including the first anchor downstream of the MSIVs.

At the maximum flow condition for power rerate (Case 1NB), the main steam piping must carry a steam flow of 15.94×10^6 #/hr, which is an increase of approximately 0.57% over the design (VWO) flow rate of 15.85×10^6#/hr. At the maximum velocity condition for power rerate (Case 3AB), the velocity in the main steam piping will increase to approximately 176 ft/sec. This velocity is within the recommended limit of 190 ft/sec. The affects of power rerate on main steam system pipe support loads due to transients (e.g., MSIV fast closure, turbine trip) are addressed in Section V, WCGS Programs Review.

Based on the preceding, it can be concluded that the existing Main Steam System piping is adequate to support power rerate.

2. Main Turbine (AC)

The function of the main turbine system is to convert the thermal energy delivered from the main steam system to mechanical energy to drive the main generator. In addition, extraction steam and condensate from the main turbine are used for feedwater heating. The system also provides a portion of the steam supply to drive the main feedwater pump turbines. The major components of the main turbine system were reviewed to determine their capability to support plant operation at the proposed power rerate condition, as follows.

Turbine-Generator

The turbine generator consists of one double flow, high-pressure (HP) and three double flow, low-pressure (LP) turbines with a rated power output of 1192.5 MWe. The GE VWO heat balance indicates that the turbine-generator will produce 1233 MWe at the VWO condition, which corresponds to an NSSS power input of 3562 MWt.

The turbine-generator was evaluated using the "Performance Evaluation of Power System Efficiencies" (PEPSE) computer program to predict turbine cycle performance at the proposed power rerate conditions. The basic input parameters used are shown in Table III-1. Actual vendor equipment data for major BOP components such as the main condenser, LP and HP feedwater heaters, condensate, heater drain and main feedwater pumps were incorporated into the model to estimate plant performance at the power rerate conditions. In addition, measured operating data such as main steam line pressure drop, moisture separator - reheater (MSR) temperature differences and an assumed 1% steam generator blowdown were used in Cases 1AB, 2AB and 3AB. A summary of the results of the PEPSE program turbine-cycle heat balance estimates are provided in Table IV.A-2. The results of the turbine - cycle heat balance estimates indicate the existing WCGS Turbine - Generator will support power operation at the proposed power rerate conditions. However, due to volumetric flow limitations of the main turbine, it is not expected the main turbine will accept the power rerate main steam flow at a pressure corresponding to Case 3NB, without possible hardware modifications. The Power Rerate conditions evaluated here will allow for a Power/Temperature coastdown, and if hardware changes are necessary and made, then it will allow operation at 104.5% power and 15 ^{o}F T_{HOT} reduction.

The bases for the turbine overspeed protection system and corresponding probability analyses are not adversely impacted by the proposed power rerate. The most limiting steam generator outlet steam flow for power rerate (Case 1NB) is 15.94×10^6 #/hr., which is only 0.57% greater than the VWO main steam flow rate. For the target operating condition, Case 2AB, the increase over the VWO main steam flow is ~0.2%. The VWO condition was assumed as the initial plant condition in the sequence of events considered for the design of the turbine overspeed protection system. Therefore, the bases for the turbine overspeed protection system is not adversely impacted by power rerate.

Moisture Separator Reheaters (MSRs)

The function of the MSRs is to improve the quality of the HP turbine exhaust steam (cold reheat) before it enters the low-pressure turbines. The MSR operation mechanically removes moisture, then reheats the dry steam using two stages of reheating. The MSRs were originally supplied by GE based on the VWO flow conditions.

The MSRs were evaluated as a part of the PEPSE program turbine-cycle heat balance calculations to predict the new reheat steam conditions associated with the rerate cases. Plant performance data at full (100%) power was used to estimate reheater terminal temperature differences (TTD), and turbine-cycle performance at power rerate conditions. The calculated reheater terminal temperature differences (TTD) for Reheater No.s 1 and 2, are shown in Table IV.A-3.

TABLE IV. A-2
SUMMARY OF WCGS POWER RERATE TURBINE CYCLE HEAT BALANCE RESULTS

CASE	DESCRIPTION	T_{HOT} REDUCTION(F)	STEAM FLOW (10^6 #/HR)	STEAM GENERATOR PRESSURE (psia)	NSSS OUTPUT (MWt)	TURBINE GENERATOR OUTPUT (MWe)	MWe CHANGE	GROSS HEAT RATE (BTU/Kw-Hr)	% STEAM GENERATOR TUBE PLUGGING
0	Base Case @ 100% Reactor Power	N/A	15.14	1000	3425	1185.8	N/A	9854	N/A
VWO[1]	VWO (105%) Main Steam Flow	N/A	15.85	1000	3562	1233.6	47.8	9852	N/A
1NB[2]	Rerate, w/o Steam Generator Blowdown	0	15.94	970	3582	1228.9	43.1	9947	0
1AB[2]	Rerate, with 1% Steam Generator Blowdown	0	15.90	970	3579	1227.8	42.0	9947	0
2NB[3]	Rerate, w/o Steam Generator Blowdown	5	15.89	908.7	3577	1213	27.2	10066	0
2AB[3]	Rerate, with 1% Steam Generator Blowdown	5	15.88	908.7	3579	1213.8	28.0	10062	0
3NB[4]	Rerate, w/o Steam Generator Blowdown	15	15.83	807	3570	1180.8	-5.0	10317	10
3AB[4]	Rerate, with 1% Steam Generator Blowdown	15	15.85	807	3579	1186.0	0.2	10298	10

NOTES:

1. Validation case based on GE VWO heat balance and plant component design and operating data.
2. "NB" cases assumed no steam generator blowdown; "AB" cases assumed 1% steam generator blowdown, and plant operating data, extrapolated to power rerate conditions.
3. Power Rerate Target Operating Condition.
4. Due to volumetric flow limitations of the main turbine, it is not expected that the main turbine will accept the power rerate main steam flow at the pressure corresponding to Case 3, without hardware modifications. Therefore, neither operation at 104.5% NSSS power, nor an uprate in electrical power output is expected to be achievable at the Case 3 power rerate conditions.

Table IV.A-3

MOISTURE SEPARATOR-REHEATER TERMINAL TEMPERATURE DIFFERENCES

CASE[1]	TTD (°F)[2] REHEATER NO. 1	TTD (°F)[2] REHEATER NO. 2
VWO	30.4	28.3
1NB	31.7	29.6
1AB	21.9	50.0
2NB	34.2	31.8
2AB	23.9	50.0
3NB	39.3	36.6
3AB	26.9	50.0

1. See Table IV.A-2 for a description of the power rerate cases.
2. The TTDs for the "AB" cases were conservatively estimated based on WCGS operating data.

The MSRs are protected from overpressurization by shell side relief valves V-935 (MSR 1B), 940 & 941 (MSR 2B), 981 & 984 (MSR 2A) and 989 (MSR 1A). The combined relieving capacity of the shell side relief valves must meet or exceed the total estimated reheat steam flow for the most limiting power rerate case at ≤110% of the MSR shell side design pressure of 270 psig. The combined relieving capacity of the shell side relief valves exceeds the maximum total reheat steam flow of 11.33×10^6 lbm/hr at the most limiting power rerate condition, Case 3. Therefore, the MSRs are adequately protected from overpressurization for the most limiting power rerate case, and based on the results of the PEPSE heat balance calculations as listed in Table IV.A-2, are adequate to support power rerate.

Main Generator

The WCGS main generator is rated at 1,373,100 KVA at a power factor of 0.9, which corresponds to a gross electrical output of (1373.1 X 0.9) = 1236 MWe. Therefore, the main generator has adequate capacity to support power rerate to a reactor power level of 3565 MWt, which corresponds to an estimated electrical output of 1229 MWe, at the maximum power rerate condition, Case 1NB, or 1214 MWe at the target operating condition, Case 2AB.

3. Condensate (AD)

The condensate system functions to pre-heat and supply condensed steam from the condenser hotwell to the steam generator feed pumps (SGFP). The condensate system includes one multi-pressure, deaerating, condenser, condensate piping, three (3) condensate pumps and four (4) stages of low pressure feedwater heating. The major condensate system components were evaluated to determine their adequacy to support the proposed power rerate to the 3565 MWt reactor power level as follows:

Main Condenser

The main condenser is a multi-pressure, deaerating unit, consisting of one high pressure (HP) shell, one intermediate pressure (IP) shell, and one low pressure (LP) shell. The main condenser functions as the steam cycle heat sink and has a design condensing duty of 7.87 X 10^9 Btu/hr. During normal operation, the condenser receives and condenses LP turbine exhaust steam, and SGFP turbine exhaust steam flows. The condenser is also a collection point for steam cycle miscellaneous flows, drains, and vents. The condenser was originally designed to accommodate a 50% load reduction without reactor trip by accepting 40% of the VWO main steam flow via the turbine bypass system. A condenser high back pressure sensor is provided to actuate an alarm at approximately 5 inches Hga and initiate a main turbine trip at approximately 7.5 inches Hga.

The main condenser was modeled in the PEPSE turbine - cycle heat balance program design mode, using condenser design data, a circulating water design inlet temperature of 80°F, and Heat Exchange Institute (HEI) standards methodology.

At the most limiting power rerate condition, Case 3NB, the condenser duty decreases to 7.842 x 10^9 Btu/hr, which is approximately 0.35% less than the VWO condenser duty of 7.87 x 10^9 Btu/hr. In addition, the capability of the main condenser to accept and condense steam following a 50% load reduction from full (power rerate) load was verified. Further, the original sizing basis of the Condenser Air Removal (CG) System remains valid for power rerate. Therefore, the main condenser and the Condenser Air Removal (CG) System are adequate for power rerating to the proposed 3565 MWt reactor power level.

Condensate System Piping

The condensate piping delivers condensate from the main condenser to the suction of the SGFPs through the condensate demineralizers and LP feedwater heaters. The condensate system piping was designed to handle 50% greater than full load flow to permit full feedwater flow following a loss of heater drain pump flow.

At the most limiting power rerate condition, Case 3, the highest velocity in the condensate pump discharge header will increase to 18.5 ft/sec. This velocity is well within the recommended design limit of 30 ft/sec. In general, condensate system pressures and temperatures do not change appreciably from the design, VWO conditions. Therefore, the condensate system piping is adequate for power rerating to the proposed 3565 MWt reactor power level.

Condensate Pumps

Three 1/3-capacity condensate pumps are provided, piped in parallel, to supply the condensed steam from the condenser through the condensate demineralizer and LP heaters to the SGFPs. The condensate pumps were designed to satisfy the full VWO feedwater flow requirements and provide 125% of the net positive suction head (NPSH) required by the main feedwater pumps during all steady state operating conditions.

At the most limiting power rerate case, Case 3AB, the total condensate flow requirement increases by approximately 6% above the VWO flow condition to 11.49x 10^6 #/hr. At this condition, the condensate pumps require approximately 8 ft of NPSH and 3000 bhp. The NPSH available for the condensate pumps at this condition is approximately 27 ft and, therefore, adequate to support condensate pump operation. At the most limiting transient case, which is a 50% instantaneous electrical load reduction accompanied by a loss of the heater drain pumps, the condensate pumps can still supply 96% of the total feedwater flow as required; the electrical load requirement for this case is approximately 3400 bhp, which is within the condensate pump motor nameplate rating of 3500 bhp. Considering the power rerate condensate flow requirements, and, that the original NPSH design margin requirements for the condensate and main feedwater pumps are satisfied, the existing condensate pumps are adequate to support power rerating to the proposed 3565 MWt reactor power level.

Low Pressure Feedwater Heater No.s 1, 2, 3, and 4

Three parallel, 1/3-capacity strings of four LP feedwater heaters are provided to heat the condensate. The LP heaters were originally sized to support plant operation with one string out of service, the condensate bypass valve passing 1/3 of VWO condensate flow, and extraction and drain flows equal to their corresponding VWO flows. LP feedwater heater performance at power rerate conditions was estimated by modeling the heaters in the PEPSE program demand mode. In addition, the feedwater heaters were modeled in detail to determine changes in terminal temperature difference (TTD) and drain cooler approach (DCA) temperature differences based on the revised flow rates and duties.

LP feedwater heater tubeside velocities at the most limiting power rerate case, Case 3 remain within Heat Exchange Institute guidelines for feedwater heaters. In addition, a review of the design tube side and shell side flow rates against the most limiting power rerate flows indicates that the power rerate flows are bounded by the original LP heater design flow rates. Based on the power rerate feedwater flow requirements, extraction steam conditions and LP heater design data, the terminal temperature difference (TTD), drain cooler approach (DCA) and heat transfer duties of the existing LP feedwater heaters have been estimated and are shown in Table IV.A-4.

Table IV.A-4

LP FEEDWATER HEATER DESIGN AND CALCULATED PERFORMANCE DATA

LP Heater No.	#1	#2	#3	#4
TTD/DCA (^{o}F)				
Design	5.00/10.0	5.00/10.0	5.00/10.0	5.00/10.0
Case 1	5.2 /10.4	5.1 /10.2	5.1 /10.1	5.2 /10.0
Case 2	5.3 /10.5	5.2 /10.2	5.1 /10.1	5.2 /10.0
Case 3	5.5 /10.8	5.4 /10.4	5.3 /10.1	5.5 /10.2
Duty (Btu/hr)				
Design	2.06×10^8	1.26×10^8	2.43×10^8	1.27×10^8
Case 1	2.09×10^8	1.28×10^8	2.46×10^8	1.29×10^8
Case 2	2.11×10^8	1.30×10^8	2.48×10^8	1.30×10^8
Case 3	2.18×10^8	1.33×10^8	2.52×10^8	1.33×10^8

where:

TTD = (steam inlet saturation temp. - feedwater outlet temp.)
DCA = (drain outlet temp. - feedwater inlet temp.)

Based on the above TTDs and heating by the HP feedwater heaters, the final feedwater temperature was calculated and the results are consistent with the temperatures provided in Table III-1 for each of the power rerate conditions. Therefore, the existing LP feedwater heaters are adequate to support power rerating.

LP Feedwater Heaters #3 and 4 Shell Side Relief Valves

The shell side relief valves for LP feedwater heaters #3 and 4 were originally sized as shown in Table IV.A-5.

Table IV.A-5

LP FEEDWATER HEATERS 3 AND 4 SHELL SIDE RELIEF VALVE DESIGN DATA

	Heater #3	Heater #4
Capacity, #/hr	450,000	388,500
Design/Set Pressure, psig	75	120
Design Temperature, °F	400	400
% Allowable Overpressure	10	10

Based on a review of the LP feedwater heater conditions at the most limiting power rerate case, the design tube side and shell side flow rates bound all power rerate flow rates. Therefore, it is concluded that the shell side relief valves for LP feedwater heaters #3 and 4 are adequately sized to protect the heaters from overpressurization for power rerating.

4. Main Feedwater (AE)

The main feedwater system delivers feedwater at the required temperature and pressure from the condensate system to the four steam generators. The feedwater system includes feedwater piping, steam generator feed pumps, isolation and control valves, high pressure feedwater heaters and flow transmitters. Each major system component was evaluated to determine their adequacy for power rerating to the proposed 3565 MWt reactor power level as follows:

Feedwater Piping

Feedwater is supplied to the four steam generators by four 14" diameter carbon-steel lines. Each line is anchored at the containment wall and designed with sufficient flexibility to provide for relative movement of the steam generators due to thermal expansion. At the most limiting power rerate condition (Case AB), the feedwater flow will increase by less than 1.5% above the VWO flow of 15.85 X 10^6 #/hr to 16.08 X 10^6 #/hr. The highest velocity in the feedwater piping at the most limiting power rerate condition is approximately 28 ft/sec., which is within the recommended design limit of 30 ft/sec. For power rerate, only the Main Feedwater System temperatures increase, as compared to the full (100%) power or VWO conditions; the process temperatures for all other plant systems remain the same, or decrease slightly. Since the original MFW System piping stress analyses utilized a feedwater temperature for thermal analyses exceeding the estimated power rerate value of 446^{o}F, the proposed power rerate will have no affect on existing MFW System stress analyses.

Steam Generator Feedwater Pumps and Turbine Drivers

Two 2/3-capacity turbine-driven steam generator feedwater pumps (SGFPs), piped in parallel, are provided to supply pre-heated feedwater to the steam generators. At the most limiting power rerate condition (Case 1AB), the required feedwater pump flow increases to 16.08 x 10^6 #/hr. This represents an increase of less than 1.5% over the VWO feedwater flow rate of 15.85x10^6#/hr. At this condition, the SGFPs require approximately 121 psi of NPSH. Based on a hydraulic review of the main feedwater system, there will be approximately 312 psi of NPSH available to the suction of the SGFPs and, therefore, an adequate margin to support SGFP operation.

The Main Feedwater Pump Turbines (MFPT) were incorporated into the turbine-cycle heat balance calculations. The brake horsepower required to drive the SGFPs is directly proportional to the SGFP flow and the total dynamic head required. Due to the reductions in steam generator pressures at the power rerate conditions, as compared to the 100% power condition, in combination with the increase in flow rate, the brake horsepower required from the MFPTs to drive the SGFPs will not change significantly. The existing SGFP Turbines are adequately sized to drive the main feedwater pumps at the proposed power rerate conditions.

HP Feedwater Heaters # 5, 6, and 7

Two parallel strings of three HP heaters are provided to heat the feedwater for supply to the steam generators. The HP feedwater heaters were originally sized based on 120% of VWO flow plus margins for fouling. HP feedwater heater # 5 can accommodate the additional duty and extraction steam/drain flows associated with one train of LP heaters out of service and 1/3 of the VWO condensate flow through the LP heater bypass valve.

At the most limiting power rerate condition (Case 1AB), the total feedwater flow requirem[illegible] 08 x 10^6 #/hr will increase by less than 1.5% above the VWO flow of 15.85x1[illegible] thus the HP feedwater heater tube side velocities and pressure drop wil[illegible] slightly. However, the feedwater heater tube side velocities are still within HE[illegible]s. The increase in feedwater heater tube side pressure drop has also bee[illegible] for in the evaluation of the main feedwater pumps and found to be acceptable. [illegible] addition, a review of the design tube side and shell side flow rates against the most limiting power rerate flow rates indicates that the power rerate flows are bounded by the design flow rates. The duty on the HP feedwater heaters will change due to the heating steam and feedwater flow requirements at power rerate conditions. The HP feedwater heaters were modeled in the PEPSE program demand mode. Based on the rerate HP feedwater flow requirements, extraction steam conditions, and HP feedwater heater design data, the terminal temperature difference (TTD), drain cooler approach (DCA) and heat transfer duty of each of the existing HP feedwater heaters were estimated, and are summarized in Table IV A-6.

Table IV.A-6

HP FEEDWATER HEATER DESIGN AND CALCULATED PERFORMANCE DATA

HP Heater No.	# 5	# 6	# 7
TTD/DCA (F)			
Design	5.00/N/A	5.00/10.0	8.00/10.0
Case 1	4.9/N/A	5.1/10.2	8.2/9.8
Case 2	5.0 /N/A	5.2/10.2	8.2/9.7
Case 3	4.8 /N/A	5.1/10.1	8.2/9.7
Duty (Btu/hr)			
Design	$2.46X10^8$	$3.26X10^8$	$3.35X10^8$
Case 1	$2.29X10^8$	$3.52X10^8$	$3.43X10^8$
Case 2	$2.34X10^8$	$3.54X10^8$	$3.45X10^8$
Case 3	$2.42X10^8$	$3.57X10^8$	$3.46X10^8$

where:
TTD = (steam inlet saturation temperature - feedwater outlet temperature)
DCA = (drain outlet temperature - feedwater inlet temperature)

Based on the above TTDs, the final feedwater temperature was calculated and the results are consistent with the temperatures provided in Table III-1 at the rerate conditions.

Based on the preceding, it can be concluded that the HP feedwater heaters are adequate to support power rerating.

HP Feedwater Heaters #5, #6 and #7 Shell Side Relief Valves

As noted above, the HP design Feedwater Heater shell side and tube side flow rates envelope the power rerate flow rates. Therefore, the existing shell side relief valves are adequately sized to protect the HP feedwater heaters from overpressurization at the proposed power rerate conditions.

Main Feedwater Isolation Valves (MFIVs)

One MFIV is installed in each of the four main feedwater lines to isolate the safety-related portions from non-safety-related portions of the system and to prevent uncontrolled blowdown from more than one steam generator in the event of a feedwater line rupture.

The MFIVs were reviewed based on the flow rates and process pressures at the most limiting power rerate condition (Case 1AB). The MFIVs were originally designed based on a mass flow rate approximately 3-1/2 times the VWO flow of 15.85×10^6 #/hr in the forward direction. Since the increase in feedwater flow to 16.08×10^6 #/hr is small (<1.5%) and the required feedwater pressure decreases for all of the power rerate cases, power rerate will have no impact on the capability of the existing MFIVs to perform their intended function. Therefore, it is concluded that the MFIVs are adequate to support power rerate.

Main Feedwater Control Valves (MFCV)

The MFCVs, in conjunction with main feedwater pump turbine speed control, provide for adjustment of steam generator water level. For the most limiting power rerate condition (Case 1AB), the feedwater flow requirement through each MFCV will increase by less than 1.5%, as compared to the VWO (design) case, and the feedwater pressure required to the steam generators will decrease. The flow increase will require slightly more pressure drop across each MFCV during power rerate operations. However, the valves have been reviewed, and are adequately sized to support operation at the power rerate conditions.

Main Feedwater Flow Meters (FE-510/FT-510 & 511, 520, 530, 540)

Two feedwater flow transmitters are provided across each feedwater flow element located in each main feedwater supply line to transmit input signals to the feedwater control system for feedwater control valve throttling. Based on a review of the instrument ranges of the flow transmitters, as compared with the most limiting power rerate conditions (Case 1AB), the existing instruments have sufficient range to function properly for power rerating.

5. Feedwater Heater Extractions, Drains and Vents (AF)

The feedwater heater extraction, drains and vents system supplies turbine extraction steam and moisture separator-reheater scavenging steam and drains to the feedwater heaters for feedwater heating, returns condensate drained from the feedwater heaters to the cycle, and provides for continuous removal of non-condensable gases from the shell side of the feedwater heaters. The system consists of the piping, valves, drain tank, pumps, controls and instrumentation which supply, drain and vent the shell side of the closed feedwater heaters. The major components of the feedwater heater extraction, drains and vents system were evaluated to determine their capability to support plant operation at the proposed power rerate condition, as follows.

Feedwater Heater Level Control and Bypass Valves

The feedwater heater level control and bypass valves were reviewed to confirm that they are adequately sized for the proposed power rerate conditions. The existing extraction, drain, and dump design flows for the WCGS feedwater heaters are based on General Electric documents "Recommended Feedwater Heater Drain Sizing Information for Nuclear Reheat Turbine-Generators". The design flow values generally exceed the values shown in the VWO flow heat balance. This additional margin is to account for unknowns in design calculation procedures and as-built plant equipment and configurations. Based on a review of the existing valve design parameters and the results of heat balance calculations made at each of the proposed power rerate conditions, the power rerate heater drain flows are lower than for the existing design case with margin. Therefore, there will be no limitations to power rerating imposed by the existing level control valves or instrumentation.

Moisture Separator Drain Tank, First Stage Reheater Drain Tanks, and Second Stage Reheater Drain Tanks

Based on a review of the results of heat balance calculations at each of the proposed power rerate conditions, the drain flows from the moisture separator reheaters to the Moisture Separator Drain Tanks, and First and Second stage Reheater Drain Tanks are lower than the design flows for the tanks, piping and valves for the system. Therefore, the existing capacities (and corresponding retention times) of the moisture separator drain tanks, first stage reheater drain tanks, and second stage reheater drain tanks are adequate to support the proposed power rerate conditions.

First Stage Reheater, Second Stage Reheater and Moisture Separator Drain Tank Level Control and Bypass Valves

The drains from each moisture separator reheater are collected in separate fir -stage reheater, second-stage reheater and moisture-separator drain tanks. The condensate drains from first stage, second stage, and moisture separator drain tanks are directed to Feedwater Heaters 6 and 7, and the heater drain tank, respectively. Level control valves on the drain lines automatically maintain the normal water levels in the drain tanks. A bypass valve is provided on each drain tank to dump the condensate to the main condenser on high level in the drain tank.

The existing level control and bypass valves were sized and furnished by General Electric. The design capacities are equal to, or exceed the VWO values. This additional margin was included to account for unknowns in design calculation procedures and as-built plant equipment and configurations. In addition, the first and second stage reheater drain tank bypass valves were sized by GE with additional capacity to accommodate the unlikely event that a CIV is closed while the feedwater heater which normally receives reheater condensate is temporarily out of service.

Based on a review of the existing valve capacities and the results of the turbine-cycle heat balance calculations for each of the proposed power rerate conditions, it can be concluded that the existing level control and bypass valves are adequately sized to support the proposed power rerate.

Heater Drain Tank Pumps, Level Control and Bypass Valves

The heater drain pumps were originally sized based on 110% of the VWO heater drains flow rate to allow for surges. The design of the heater drain pump system is based on providing at least 25 percent excess pump NPSH at the design (VWO) flow condition.

The vendor's heater drain pump performance curve was used in the turbine cycle heat balance calculations to determine the pump operating point at each of the proposed power rerate conditions.

Based on the results of the power rerate heat balance calculations, the total estimated heater drain pump flow increases by approximately 4% as compared with the VWO flow, and remains within the capabilities of the existing heater drain pumps. The adequacy of the heater drain pump control valves was reviewed at each of the proposed power rerate conditions. The heater drain pump control valves are adequately sized for power rerate Cases 1, 2 and 3, as listed in Table III-1.

6. Condensate Demineralizer (AK)

The Condensate Demineralizer System (CDS) is designed to maintain the required purity of feedwater for the steam generators. The system removes corrosion products, suspended solids and impurities entering the condensate system from condenser leakage by filtration and ion exchange. The CDS consists of six demineralizer vessels piped in parallel; with up to five vessels in operation and one vessel in standby.

The service for each demineralizer vessel is terminated on either high differential pressure across the vessel (approximately 25 psid) or high cation or sodium content in the demineralizer effluent. The CDS performs no safety function.

The CDS system was sized based on the VWO condensate flowrate of 21,600 gpm, and is designed to accommodate flow surges as high as 144% of VWO condensate flow (31,200 gpm) during transient conditions. For the target operating power rerate case, Case 2AB, the condensate flow of 22,800 gpm through the CDS exceeds the VWO flowrate of 21,600 gpm by less than 6%, and the pressure differential across each demineralizer vessel is estimated to increase by less than 4 psi. This will result in a slight reduction in the service run time of each demineralizer vessel, based on maintaining the existing differential pressure alarm setpoint. Based on input from the CDS original equipment manufacturer (OEM), although the power rerate condensate flows exceed the (continuous) flow rate for which the condensate demineralizer has been designed, this small increase in flow, and the corresponding increase in pressure drop can be accommodated by the CDS.

In addition, the differential pressure indicating switches (PDIS) for the CDS vessels have an adjustable setpoint range of 0-50 psid, and there is adequate NPSH available for the safe operation of the steam generator feedwater pumps (SGFP) should it be necessary to raise the differential pressure setpoint to account for this increase in flowrate, and thereby restore the CDS vessel run time (if pressure drop is limiting). Based on the preceding, it is concluded that the CDS is adequate to support the power rerate.

7. Demineralized Water Storage and Transfer (AN)

The demineralized water storage and transfer system (DWSTS) receives filtered, demineralized water from the demineralized water makeup system (DWMS), and stores and transfers demineralized water to plant systems and components on demand. A review of the systems and components supplied from the DWSTS (see Table IV.A-7) indicates that there will not be a significant increase in demand or consumption of demineralized water for power rerate. Therefore, the existing DWSTS is adequate to support power rerate.

Table IV.A-7

DEMINERALIZED WATER STORAGE AND TRANSFER SYSTEM LOADS

Condensate Storage Tank
Reactor Makeup Water Storage Tank
Component cooling water system (surge tank, chemical addition tank)
Closed cooling water system
Auxiliary steam system (chemical addition & cond. recovery)
Diesel Generator cooling water expansion tank
Chilled water system (expansion tank)
Hot water system
Miscellaneous laboratory and sampling requirements
Miscellaneous flushing requirements
Miscellaneous makeup requirements
Condensate pump seals
Condensate and chemical addition
Nuclear sampling room
Hot machine shop
Radwaste and chemical laboratories
CVCS Positive Displacement process sampling system
Feedwater Chemical Addition System
Solid radwaste resin sampling station
Stator cooling water system
Condenser in-leak detection rack
Cask Washdown Pit
CVCS Chiller Surge Tank
Condensate Polishing System
Secondary Liquid Waste Evaporator

8. Condensate Storage and Transfer (AP)

The condensate storage and transfer system (CSTS) serves as a reservoir to (1) supply and receive condensate as initiated by the condenser hotwell level control system, (2) provide adequate inventory for initial fill of the condensate and feedwater systems, the steam generators and the condenser hotwell, and (3) retain a minimum usable inventory to support the operation of the auxiliary feedwater system (AFWS) to maintain the plant at hot standby for 4 hours, followed by plant cooldown to an RCS temperature of 350°F at 50°F/hour. The original sizing basis of the CST was a reactor thermal power of 3565 MWt.

Analysis has indicated the minimum condensate inventory necessary to satisfy the AFWS make-up requirement for power rerate remains less than 281,000 gallons, as currently defined by WCGS Technical Specification and Basis 3.7.1.3, to maintain the AP System operable. No significant changes in the make-up requirements to the condensate or feedwater systems are required as a result of power rerate. Therefore, the existing condensate storage and transfer system is adequate to support power rerate to the proposed 3565 MWt reactor power level without modification.

9. Circulating Water (DA)

The circulating water system (CWS) provides the cooling water flow to the main condensers for the removal and rejection of waste heat to the normal plant heat sink. The CWS sizing basis was the VWO heat balance for WCGS, a maximum design supply temperature of 80°F, and a temperature rise of 30°F. The circulating water design flow rate of 530,000 gpm was used as a basis for evaluating the adequacy of the main condenser, as part of the power rerate turbine-cycle heat balance calculations. The calculated circulating water temperature rise, based on a supply temperature of 80°F, across the main condenser is 30.2°F for Case 1AB, 30.4°F for Case 2AB, and 30.8°F for Case 3AB. Also, actual operating information from Wolf Creek and other power plants note that a 40 °F rise in temperature across the condenser does not create an unreviewed environmental question. That is, it will not significantly increase previously evaluated impacts, change effluents, or constitute an impact not previously reviewed. Based on the preceding, it can be concluded that the Circulating Water System is adequate to support power rerate to the 3565 MWt reactor power level.

B. Reactor and Steam Generator Controls and Auxiliary Systems

1. Reactor Makeup Water (BL)

The reactor makeup water system (RMWS) receives filtered, deaerated demineralized water from the demineralized water storage and transfer system and from several systems that process waste water to be recycled within the plant. The reactor makeup water system stores the water to be used in the plant on demand for primary makeup water requirements. The RMWS has no active safety design basis.

A review of the systems and components supplied by the RMWS indicates that there will be no significant changes in Reactor Makeup Water demand or consumption as a result of power rerate. Therefore, the existing reactor makeup water system is adequate to support power rerate to the proposed 3565 MWt reactor power level.

2. Steam Generator Blowdown (BM)

The Steam Generator Blowdown System (SGBS) is designed to maintain the steam generator secondary side water chemistry within the specifications prescribed by the NSSS Supplier while recovering heat from the blowdown and treating the blowdown for return to the condenser.

Portions of the SGBS have safety design bases such as remaining functional following a DBA or a postulated hazard, withstanding the effects of natural phenomena, and maintaining the isolation capability of the secondary side of the steam generators.

The SGBS is designed to ensure treatment of up to 176,000 lb/hr (44,000 lb/hr per steam generator) during power generation, and maintain the plant effluent within the radiological specifications for plant discharge during abnormal operation with primary-to-secondary steam generator leakage. The SGBS flow rates are adjusted from approximately 30,000 lb/hr to 130,000 lb/hr to match the variable nature of secondary water chemistry. Steam generator blowdown flow rates are a function of the plant steaming rate, which will increase approximately 5% over the current, 100% power condition due to power rerating. Since the blowdown rate is administratively controlled based on the steaming rate and the secondary side water chemistry, the only significant change in the SGBS operation due to power rerate will be that, due to the reduction in steam generator pressure, the blowdown control valves will have to be throttled further open to achieve a higher blowdown flow rate, as compared to the current (100%) power condition. However, adequate margin exists in the blowdown flow control valves to accommodate the pressure decrease and increased blowdown flow rate at the target operating condition (Case 2AB). Therefore, the SGBS is adequate to support operation at the proposed power rerate condition.

3. Borated Refueling Water Storage (BN)

The borated refueling water storage system receives and stores borated, demineralized water to (1) flood the refueling pool during refueling (2) supply the CVCS during abnormal operating conditions and (3) supply the Containment Spray and ECCS during accident conditions. The volume required for refueling and for accident conditions (CSS and ECCS) will not change due to power rerating and the minimum boron concentration will not be revised. Therefore, the existing level settings in the Refueling Water Storage Tank do not require revision and the existing RWST volume is adequate to support power rerating to the proposed 3565 MWt reactor power level.

4. Auxiliary Feedwater (AL)

The auxiliary feedwater system (AFS) is a reliable source of water for the steam generators. The auxiliary feedwater system provides a safety grade water supply following a secondary side line rupture or loss-of-offsite power. The auxiliary feedwater flow will ensure adequate makeup to the steam generators to prevent the reactor coolant pressure from increasing and causing release of reactor coolant through the pressurizer relief and/or safety valves.

The auxiliary feedwater system may also be used following a reactor shutdown in conjunction with the condenser dump valves or atmospheric relief valves to cool the reactor coolant system to 350 ^{O}F and 400 psig, at which temperature the residual heat removal system is brought into operation. The normal source of feedwater for plant start-up and shutdown is the motor-driven feedwater pump.

A review of the actuation scenarios for the AFS was performed noting that all affects on the AFS, as a result of rerate, are initiated at the Steam Generators (S/G).

1. The power rerate effects on the secondary side are to increase main feedwater temperatures and mass flowrates, decrease outlet steam temperatures and pressures, and increase the mass flowrate of the outlet steam. Upon a trip, the AFS will generally encounter equal or lower S/G temperatures and pressures.

2. Per analysis the current minimum required volume of water in the Condensate Storage Tank (CST), noted in the Tech Spec 3/4 (3.7.1.3) as 281,000 gallons, is sufficient to cool down the plant to allow operation of the Residual Heat Removal System. Therefore, no additional flow is required.

3. The Main Feedline Break accident scenario (the most limiting break accident for the AFS) was remodeled incorporating rerate conditions along with updating the analysis. Results show the Main Steam feed to the AFS Turbine Driven Feed Pump will remain at a higher pressure than earlier bounding analyses.

Therefore, the AFS is adequate for operation at all rerate conditions.

C. Turbine Generator Control and Auxiliary Systems

The main generator is rated at 1.373 x 10^6 KVA at a power factor of 0.90, corresponding to a gross electrical output of 1236 MWe. Under the proposed target operating condition for power rerate, Case 2AB, the main generator output is estimated to be approximately 1214 MWe.

The Turbine Generator Control and Auxiliary Systems were reviewed to identify any limitations which may exist within their design which would prevent them from supporting operation at the proposed target power rerate operating condition. Based on this review, it was determined that each of the systems was originally designed based on the performance requirements dictated by the VWO heat balance. As stated above, the proposed power rerate target operating condition does not exceed the maximum rating of the generator. Therefore, the Turbine Generator and Auxiliary Systems listed in Table IV.C-1 are adequate to support operation at the proposed power rerate condition.

Table IV.C-1

TURBINE GENERATOR AND AUXILIARY SYSTEMS

Steam Seals (CA)
Main Turbine Lube Oil (CB)
Generator Hydrogen and Carbon Dioxide (CC)
Generator Seal Oil (CD)
Stator Cooling Water (CE)
Lube Oil Storage, Transfer and Purification (CF)
Main Turbine Control Oil (CH)

D. Equipment Cooling Water Systems

1. Service Water (EA)

The Service Water System (SWS) provides a source of heat removal to non-essential auxiliary plant equipment and to the Essential Service Water System (ESWS) during normal operation and normal shutdown. The system design basis is to provide a minimum flow of approximately 42,500 gpm to the power block at a maximum design temperature of 90^{o}F. The SWS has no safety design basis.

The SWS provides cooling water at a maximum temperature of 90°F to the components in Table IV.D-1 (the values in parenthesis indicate the number of components in use during full power operation).

Table IV.D-1

SERVICE WATER SYSTEM COOLING LOADS

Closed Cooling Water Heat Exchangers EEB01A,B (2)
Central Chiller Condenser Units SGB01A,B (2)
Steam Packing Exhauster ECA01 (1)
Air Compressor & After Cooler CKA01C (1)
Generator Hydrogen Coolers ECC01A,B (2)
Generator Stator Liquid Coolers ECE01A,B (2)
Turbine-Generator Lube Oil Coolers ECB01A,B (2)
Chemical & Volume Control System Chiller SBG02 (1)
Steam Generator Blowdown Non-Regenerative Hx EBM02 (1)
Condenser Vacuum Pump Seal Water Coolers ECG01A,B,C (3)
Water Box Venting Pump Seal Water Coolers EDA01A,B,C (3)

In addition, the SWS provides cooling water to the Essential Service Water System (ESWS) to remove Component Cooling Water System loads during normal operation. Based on a review of the normal operation CCW system cooling loads, there will be no increase in any CCW system cooling loads for power rerate.

The normal (100% power operation) SWS flow rate is 18,153 gpm to non-essential components and 24,282 gpm to ESWS components, for a total of 44,435 gpm, based on a service water supply temperature of 90°F.

Some of the components (Table IV.D-1) cooled by the SWS will experience slightly higher heat loads while operating at rerated power conditions, as compared to the heat loads applicable at the current 100% power condition. However, heat loads for these components were based on component design requirements corresponding to the VWO operating condition. Therefore, based on the margins designed into the SWS, the system is adequate to support power operation at the proposed power rerate condition.

2. Closed Cooling Water (EB)

The Closed Cooling Water System (ClCWS) provides cooling water to dissipate heat from the following auxiliary components required for power generation (the values in parenthesis indicate the number of components in use during full power operation):

Table IV.D-2

CLOSED COOLING WATER SYSTEM COOLING LOADS

Generator isophase bus duct coolers, EMA01 & EMB01 (1)
Steam generator feed pump turbine lube oil coolers, EFC01A, B, C & D (2)
Generator excitor air coolers, EMB01A (1)
EHC coolers, ECH01A & B (1)
Condensate pump motor bearing oil coolers, PAD01A, B & C (3)
Secondary system sample coolers and chiller unit, ERM01, 03-12, and 14 (11)
Heater drain pump motor bearing oil coolers, PAF01A & B (2)
Steam generator wet layup system sample coolers, EBM03A, B, C & D (4)
Condensate demineralizer chiller unit, EAK04 (1)
Auxiliary boiler sample cooler, EFA02 (1)
Auxiliary steam reboiler sample, FB267 (1)
Degasifier vacuum pump, CAN01A & B (1)

The normal (100% power operation) ClCWS flowrate is 442 gpm which corresponds to a heat duty of 3.5×10^6 BTU/Hr. The design heat removal capability of the ClCWS is 7.56×10^6 BTU/Hr with a system design flowrate of 1062 gpm.

The equipment heat loads used for design of the ClCWS were based on the VWO rating of the main Turbine-Generator (1236 MWe). The component heat loads for power rerate are not expected to significantly exceed the rated values for these components. Therefore, the present ClCWS flowrates are judged to be adequate to support power rerate.

The first four components listed in Table IV.D-2 constitute approximately 89% of the total system heat duty, and are provided with dedicated automatic control valves at the heat exchanger return connections. These control valves are automatically throttled in response to temperature signals at the outlet of the equipment being cooled, thereby ensuring proper flowrates. Based on the preceding, it can be concluded the Closed Cooling Water System is adequate to support power rerate.

3. Essential Service Water (EF)

The Essential Service Water System (ESWS) consists of two redundant cooling water trains and provides cooling water to plant components requiring cooling for safe shutdown of the reactor following an accident.

The ESWS also provides emergency makeup to the spent fuel pool and the component cooling water system. The ESWS is the backup water supply for the auxiliary feedwater system.

New analysis of the heat inputs to/from the Component Cooling Water System and Containment Air Coolers, including power rerate changes, for normal and Post-LOCA operation determined no significant increases outside original bounding conditions or analyses. Though there is a slightly higher heat input to the ESWS when integrated over a period of time, the peak heat inputs to the ESWS are less than the current expectations.

All other heat inputs to the ESWS are unaffected by rerate. Therefore, the ESWS is adequate to fulfill its safety design functions at power rerate conditions.

Ultimate Heat Sink (UHS)

This section addresses operation at 104.5% power and it's effect on the UHS thermal performance with respect to the design basis temperature of the cooling water supplied to the plant.

The heat rejected from the plant to the UHS has been tabulated for the proposed operation at 104.5% power for the worst case (normal and/or accident) condition that would require shutdown after a main dam failure.

The minor changes in heat rejection (0.2% increase for LOCA) were evaluated for their potential to change the results of the current transient thermal analysis. The effect on the UHS outlet temperature (plant inlet) due to this increase is minor compared to the natural heating from the environment. Any change in the lake outlet temperatures for the worst condition would be insignificant, less than the accuracy of the LAKET computer program which reports results to the nearest 0.01 ^{O}F.

The current UHS thermal analysis remains valid at conditions of 104.5% power as proposed by the Power Rerate Program and conditions outlined in USAR Section 9.2.5.2.5.

4. Component Cooling Water (EG)

The Component Cooling Water System (CCWS) provides cooling water to selected nuclear auxiliary components during normal plant operation and will provide cooling water to engineered safety feature systems during a loss-of-coolant accident (LOCA). This system is a closed loop system which acts as an intermediate barrier between the Essential Service Water System (ESWS) or the Service Water System (SWS) and potentially radioactive systems.

A review of the CCWS heat loads after power rerate find that normal and accident heat loads after power rerate are bounded by pre-rerate analyses. The only significant affect of rerating on the CCWS is slightly elevated long-term heat removal requirements. However, the maximum heat removal requirement at rerate conditions is less than or equal to current heat removal demands.

5. Containment Spray System (EN)

The two functional objectives of the containment spray system (CSS) as an engineered safety feature are: 1) to reduce the containment atmosphere temperature and pressure in the event of a loss-of-coolant accident (LOCA) or a main steam line break (MSLB) inside containment and 2) to limit the offsite radiation levels in the event of a postulated LOCA. The system provides two mechanisms to meet these objectives:

a. The containment spray system delivers cold spray chemical solution water to the containment to reduce the atmospheric temperature and pressure, and thereby diminishes the driving force for leakage of fission products from the containment to the environment.

b. The containment spray solutions' chemical characteristics enhance the removal of the airborne fission products from the atmosphere. Thus, the containment spray system serves to reduce the airborne fission product inventory available for leakage.

Based on new pressure/temperature models of the LOCA and MSLB events, including new rerate conditions, the old existing analyses are still bounding.

A review of the off-site and control room dose calculations with the new rerate conditions notes that doses will not increase beyond current estimated levels in either area.

The Containment Spray System chemical addition - will be readjusted to assure sufficient NaOH is delivered to accommodate 2500 ppm boron in the RCS. This will occur because of the potential need for chemistry control of the containment sump in the event of a LOCA.

Therefore, the CSS is adequate to support power rerate.

E. Plant HVAC Systems

1. Central Chilled Water (GB)

The central chilled water system supplies cooling water to the cooling coils located in various rooms and areas of the plant to provide a suitable environment for personnel and equipment. The system has no safety design basis. Based on a review of the areas and equipment cooled by the system, the equipment and area heat loads are not expected to change due to power rerate. Therefore, the existing Central Chilled Water System is adequate to support power rerating to the proposed 3565 MWt reactor power level.

2. Fuel Building HVAC (GG)

The fuel building HVAC system provides conditioned outside air for ventilation, cooling and heating of the Fuel Building, collects and processes airborne particulates following a postulated fuel handling accident, provides a suitable ambient temperature for the SFP pump motors and provides supplemental fuel building heating. A review of the sources of heat loads, ventilation requirements, heating requirements and fission product source terms indicates that power rerating will not adversely impact this system. Therefore, the existing fuel building HVAC system is adequate to support power rerating to the proposed 3565 MWt reactor power level.

3. Auxiliary Building HVAC (GL)

The auxiliary building HVAC system provides conditioned outside air for ventilation and cooling and heating of various areas of the auxiliary building, collects and processes airborne particulates during normal operation and post-LOCA, provides a suitable ambient environment for the electric motor drives for ECCS pumps and provides supplemental building heating when required. A review of the sources of heat loads, ventilation requirements and heating requirements indicates that power rerating will not have any effect on this system. Therefore, the existing Auxiliary Building HVAC System is adequate to support power rerating.

4. Containment Cooling (GN)

The containment cooling system provides the means of cooling the containment (1) during normal plant operation to maintain a suitable environment for the contained equipment, and (2) following a LOCA or MSLB to reduce the containment temperature and pressure, and thus reduce the potential leakage of gaseous radioactivity.

(Evaluation of LOCA and MSLB for power rerating was performed and incorporated the heat removal capability of the Containment Cooling System (via ESW flow rate and inlet temperature). Therefore, only Containment Cooling during normal operation is addressed here).

The containment coolers provide cooling by recirculation of containment air across air-to-water cooling coils. The cooled air is distributed and directed to the heat sources by distribution ductwork, the CRDM cooling system, cavity cooling system, hydrogen mixing fans, pressurizer cooling fans and elevator machine room exhaust fan.

A review of the containment cooling system design heat loads was performed to determine the effects of the rerate on the heat loads and the containment cooler heat removal capabilities during normal plant operation. The existing containment heat loads can be divided into the following six major groups:

1. Electrical Equipment Heat Loads (lighting and motors)
2. Control Rod Drive Mechanisms (CRDM)
3. Reactor Coolant System (RCS) Leakage
4. Secondary System Piping Heat Losses
5. RCS Piping Heat Losses and Equipment Heat Loads
6. Solar load

The six groups above comprise the total containment cooling design heat load of approximately 9.24×10^6 BTU/hr applicable to the normal operating conditions. Of these six groups, the contributions of the Solar, Electrical Equipment, CRDM Heat Loads and RCS Leakage are unaffected by the rerate. These four groups make up approximately 67% of the existing, normal operating containment heat load.

Implementation of the power rerate will result in a slight increase in the main feedwater system operating temperature and a slight decrease in the main steam system operating temperature. An examination of the existing secondary system piping heat loads indicates that the net result of these changes will be to approximately offset each other. In addition, it should be noted that the total secondary piping heat load comprises approximately 6% of the total containment heat load. The sixth and remaining group, RCS Piping and Component Heat Loads, comprise approximately 27% of the total containment heat load. The change in RCS operating temperature as a result of rerate will change this heat load. For the power rerate target operating case,

Case 2AB, all RCS system temperatures decrease, as compared to the current design (100% power) case. Therefore, the power rerate will not have an adverse impact on the containment heat loads, and the containment cooling system is considered adequate during normal plant operation to support power rerating.

5. Containment Atmosphere Control (GR)

The containment atmosphere control system (CACS) reduces the concentration of radioiodine and particulate activity within the containment prior to and during personnel access or purging of the containment during reactor power operation. The CACS collects and processes airborne and particulate fission products through charcoal adsorbers. Prior to entrance into containment during reactor power operation, the CACS operates in conjunction with the mini-purge system to maintain operator exposure from airborne activity to less than those specified in 10 CFR 20. The CACS performs no safety function.

The effect of power rerate on the normal operating radioiodine and particulate activities within containment will be to increase the activity levels approximately in proportion to the increase in reactor power level, or ~4.5%. The effect of this increase would, at most, result in a slight increase in the frequency for particulate filter and/or charcoal adsorber replacement. Based on the original design margins incorporated into the system, and the negligible impact expected on the containment normal operating radioiodine and particulate activities due to power rerate, it can be concluded that the Containment Atmosphere Control System is adequate to support power rerating to the 3565 MWt reactor power level.

6. Containment Hydrogen Control (GS)

Power rerate would potentially impact the performance of the Containment Hydrogen Control System (GS) if it results in an increase in the hydrogen generation rate, or increases the total quantity of hydrogen in containment. The significant sources of hydrogen in containment post-LOCA include, (1) metal-water reaction, (2) radiolytic decomposition of post-LOCA emergency cooling solutions, and, (3) corrosion of metals and paints by solutions used for emergency core cooling or containment spray. The GS system is designed to maintain the containment hydrogen concentration below 4.0% by volume, post-LOCA.

The WCGS containment post-LOCA hydrogen generation analysis was reviewed to identify any impact due to power rerate. With respect to metal-water reaction, Westinghouse has indicated that the original assumption used in the hydrogen generation analysis to conservatively estimate the hydrogen evolved from the zirconium-water reaction remains applicable for power rerate (i.e., the total amount of hydrogen generated from the chemical reaction of the fuel cladding with water or steam does not exceed 1% of the hypothetical amount that would be generated if all of the metal in the fuel cladding were to react). There will be no measurable increase in hydrogen generation for power rerate due to radiolysis, since the current hydrogen generation analysis is based on a reactor core power of 3636 MWt, or 102% of the power rerate reactor core power of 3565 MWt.

Although power rerate will not add aluminum or zinc bearing materials to the WCGS containment, the applicable LOCA analysis does result in an increase in the long term containment temperature profile, due primarily to a reduction in Essential Service Water flow to the containment air coolers. Since corrosion rates increase with increasing temperature, the rate of hydrogen evolved from the corrosion of metals and paints in containment by ECCS or Containment Spray fluids, was reviewed for power rerate. The results of this review indicate that the containment atmosphere hydrogen concentration will reach 3% by volume in approximately 9.5 days, based on hydrogen evolved from the three major sources listed above, and the current inventory of aluminum, zinc and zinc-based paints in containment. This duration was calculated assuming, conservatively, no operation of the existing hydrogen recombiners. Therefore, the rate of hydrogen generation, post-LOCA, for power rerate is bounded by the current hydrogen generation analysis. It can therefore be concluded that the existing Containment Hydrogen Control System is adequate for power rerate.

7. Containment Purge (GT)

The containment purge system reduces the concentration of noble gases within the containment prior to and during personnel access during reactor power operation and maintains a suitable environment during cold shutdown. The system consists of the shutdown purge sub-system and the mini-purge sub-system. Prior to personnel entrance into containment, the mini-purge system operates to reduce the containment noble gas concentration. During cold shutdown, fresh outside air is supplied by the containment shutdown purge sub-system and exhausted by the containment shutdown purge exhaust fan via a filter adsorber unit and then vented under monitored conditions. In addition, the GT System provides an alternative means to the Containment Hydrogen Control System for reducing hydrogen gas concentrations within containment, post-LOCA. The safety design bases of the GT System include containment isolation design provisions applicable to the shutdown purge and mini-purge containment isolation valves.

As addressed in the evaluation for the Containment Hydrogen Control System, the post-LOCA hydrogen generation rate applicable to power rerate is bounded by the original analysis. The other functions of the GT System are not significantly affected by power rerate, as they are dependent upon the normal operating containment noble gas concentration, and the cold shutdown, containment ambient conditions. The only other impact that power rerate could potentially have on the GT System relates to the ability of the containment mini-purge isolation valves to close during the first few seconds of a design basis LOCA, if open. The applicable LOCA analysis was reviewed to identify any increase in containment pressures, as would be seen by the mini-purge isolation valves, either while closing, or once closed. Table IV.E-1 provides a comparison of the original design basis and power rerate LOCA containment pressures used to evaluate the ability of the purge valves to close, and remain closed.

Table IV.E-1

CONTAINMENT MINI-PURGE ISOLATION VALVE
CONTAINMENT PRESSURE AT VALVE CLOSURE AND PEAK PRESSURE
FOLLOWING A DOUBLE ENDED PUMP SUCTION GUILLOTINE BREAK LOCA

	Containment Pressure At 6 seconds (psia)	Containment Peak Pressure (psia)
Post-Power Rerate LOCA	38.89	62.05
Pre-Power Rerate LOCA	37.626	61.959
Pressure Increase	1.264 (3.36%)	0.091 (0.15%)

As shown in Table IV.E-1, power rerate will result in a small increase in the containment pressure at six (6) seconds (the time at which the mini-purge valve is fully closed), as well as the containment peak pressure, as compared to the original design basis LOCA. The capability of the containment mini-purge valves to perform their containment isolation function was evaluated in detail and documented in the SNUPPS Containment Purge Valve Operability Report. The results of this evaluation, as documented in the report, indicate that the ratio of actuator torque available to torque required varies from a minimum of 7.0 to a maximum of 14.5, depending upon valve position. In addition, it was determined that material stresses within the critical parts of the assembly remain well within allowables at the maximum loading conditions. Due to the small increase in containment pressure as a result of power rerate, and the substantial margins in excess actuator torque and valve/actuator material strengths, the containment mini-purge valves are adequate for power rerate.

F. Radwaste Systems

1. Gaseous Radwaste (HA)

The gaseous radwaste system continuously processes fission product gases removed from the reactor coolant system and intermittently from other process system sources that contain hydrogen. Westinghouse has indicated that there will be no changes in the concentration or quantities of the fission product gases due to power rerating. The WCGS fission product source terms used for the design of the Gaseous Radwaste System were based on a reactor power level of 3565 MWt. Therefore, the existing gaseous radwaste system is adequate for power rerating to the proposed 3565 MWt reactor power level.

2. Liquid Radwaste (HB)

The liquid radwaste system collects and processes potentially radioactive liquid wastes including tritiated wastes, high level chemical wastes, controlled access area floor drainage and laundry and personnel decontamination waste. Westinghouse has indicated that there will be no changes in the concentrations or quantities of the fission product in the source liquids due to power rerating. The WCGS Liquid Radwaste System fission product source terms used for the design of the Liquid Radwaste System were based on a reactor power level of 3565 MWt. Further, the Liquid Radwaste System functions as a batch process and changes in demand (if any) due to power rerate can be accommodated by changing the frequency of processing. Therefore, the existing Liquid Radwaste System is adequate for power rerating to the proposed 3565 MWt reactor power level.

3. Solid Radwaste (HC)

The Solid Radwaste System collects and packages potentially radioactive solid wastes including spent resins, evaporator bottoms, filter cartridges, chemical wastes, reverse osmosis wastes, dry wastes and compacted wastes. Westinghouse has indicated that there will be no changes in the concentrations or quantities of fission products in the source fluid systems due to power rerating. The WCGS Solid Radwaste System fission product source terms used for the design of the Solid Radwaste System were based on a reactor power level of 3565 MWt. Further, the solid radwaste system functions as a batch process and changes in demand (if any) due to power rerate can be accommodated by changing the frequency of processing. Therefore, the existing Solid Radwaste System is adequate for power rerating to the proposed 3565 MWt reactor power level.

4. Boron Recycle (HE)

The boron recycle system collects borated radioactive water from the reactor coolant system and other process system components. Borated water is processed into reusable water and boric acid for makeup to the reactor coolant system. Westinghouse has indicated that any changes in the reactor coolant system boron concentrations due to power rerating will be insignificant, and no change in the boron concentration is planned for power rerate. The boron recycle system functions as a batch process and changes in demand (if any) due to power rerate can be accommodated by changing the frequency of processing. Therefore, the existing boron recycle system is adequate for power rerating to the proposed 3565 MWt reactor power level.

5. Secondary Liquid Waste (HF)

The secondary liquid waste system processes potentially radioactive liquid wastes collected in the Turbine Building, including deep bed condensate demineralizer regenerant waste and Turbine Building floor and equipment drains. Westinghouse has indicated that there will be no changes in the concentration or quantities of the fission products in the source liquids due to power rerating. The WCGS fission product source terms used for the design of the Secondary Liquid Waste System were based on a reactor power level of 3565 MWt. The design basis allowable primary-to-secondary leakage rate will not be changed due to power rerating. Further, the secondary liquid waste system functions as a batch process and changes in demand (if any) due to rerating can be accommodated by changing the frequency of processing. Therefore, the existing secondary liquid waste system is adequate for power rerating to the proposed 3565 MWt reactor power level.

G. Main Generator and Auxiliary Systems

1. Main Generator (MA)

The function of the Main Generator (MA) System is to convert the mechanical energy produced by the Main Turbine into electrical power and transmit it through the isolated phase bus and the main step-up transformer to the off-site power system. It also serves to step down voltage through the unit auxiliary transformer for normal operation of the plant unit auxiliaries. Transient Recovery Voltage (TRV) Surge Capacitors are used to limit the TRV rise time on the downstream 13.8 KV circuit breakers. The MA system has no safety design basis.

The plant main generator is rated at 1236 MWe (1,373.1 MVA X 0.9pf) gross electrical output. Therefore, the main generator has adequate capacity to support the power rerate to a reactor power level of 3565 MWt, which corresponds to an electrical output of approximately 1229 MWe for Case 1NB, and 1214 MWe at the target operating condition, Case 2AB.

The unit auxiliary transformer has adequate capacity to supply both Group 1 and Group 2 auxiliary loads of the plant to permit full power operation. The forced air cooling equipment on the isolated phase bus has sufficient redundancy so that failure of any one active component will not require a reduction in load on the main generator.

The Main Generator system protective relay settings were reviewed to evaluate the impact of power rerating, and it was determined that no revisions are required to the present relay settings. Power rerate has no impact on the TRV surge capacitors. Based on the preceding, it is concluded that the Main Generator System is adequate to support power rerate.

Main Step-Up Transformers

The main step-up transformer consists of three (3) single phase units rated 415 MVA each. The anticipated generator output (1,365.6 MVA) after the 4.5% unit uprate showed that the capacity of the main transformers would not be adequate. WCNOC will modify the transformers to provide a new nameplate rating of 448.2 MVA. This is an increase of 8% which will result in a main transformer capability of 1344.6 MVA. The unit auxiliary transformer will consume approximately 45 MVA, therefore the gross generating capacity minus the unit auxiliary loads (1365.6 MVA - 45 MVA = 1320.6 MVA) provides a sufficient margin of capacity (1344.6 MVA - 1320.6 MVA = 24 MVA). The increase in transformer nameplate rating is accomplished by installing a larger capacity oil cooling system.

2. Excitation and Voltage Regulation (MB)

The function of the Excitation and Voltage Regulation System is to provide the source of field current for excitation of the Main Generator and to control generator voltage by controlling its excitation through voltage regulation. The MB system has no safety design basis.

The excitation and voltage regulation system permits continuous full power operation of the main generator. The excitation system has a minimum base voltage response ratio of 0.50. The performance of the regulation with one rectifier section out of service remains 100% of normal operation. The generator CT's and PT's used with the regulation system cannot be shared with any other burden.

The existing exciter and voltage regulation system has adequate capacity to support power rerate to a reactor power level of 3565 MWt, which corresponds to an electrical output of 1214 MWe, at the target operating condition, Case 2AB.

3. Startup Transformer (MR)

The function of the startup transformer system is to receive power from the offsite source and step down the voltage to supply the onsite electrical distribution system for the startup and shutdown of the nuclear generating unit. The startup transformer also serves as the source of power for one load group of the Class 1E power system. Transient Recovery Voltage (TRV) Surge Capacitors are provided to limit the TRV rise time on the downstream 13.8 KV circuit breakers.

The startup transformer is connected to an independent and redundant off-site power source and supplies the Class 1E System as one of two preferred, redundant sources. The startup transformer has adequate capacity to supply both Group 1 and Group 2 auxiliary loads to permit full power operation of the generating unit.

The startup transformer is a 3 Phase, oil filled, OA/FA/FOA, 60/80/100 MVA with H winding: 345 KV wye, 1050 KV BIL, and X/Y windings: 13.8 KV wye, 110 KV BIL double secondary. The startup transformer feeds two 13.8 KV buses and a second ESF transformer, XNB02. In the event that the generator or unit auxiliary transformer is taken out of service, the startup transformer will supply the total auxiliary system loads. If the startup transformer fails to provide preferred power to ESF transformer XNB02, the offsite power supplied by alternate source through ESF transformer XNB01 provides power to 4.16 KV Class 1E buses so that the system is not affected. The offsite power circuits, including the transformers and cables, have been sized to carry their rated loads continuously.

Power rerate will result in slight increases in the loads on the MR system as follows:

a. The four (4) Reactor Coolant Pump (RCP) motors, which are fed from PA01 & PA02, will exceed their nameplate rating (7000 HP) and cold loop rating (8750 HP) by 92 HP & 324 HP, respectively, for the most limiting power rerate condition (10% SGTP). These increases in RCP motor loads have been reviewed by Westinghouse and found to be within the capabilities of the existing motors.

b. The three (3) Condensate Pump motors, which are fed from PB03 & PB04, will drive loads slightly above their present operating loads, but below their full load nameplate rating (3500 HP).

c. The two (2) Heater Drain Pump motors, which are fed from PB03 & PB04, will drive loads slightly above their present operating loads, but below their full load nameplate rating (1500 HP).

The startup transformer protective relay settings were reviewed to evaluate the impact of power rerating, and it was determined that no changes are required to the present relay settings.

Based on the preceding, the MR System is adequate to support power rerate.

4. Switchyard (S4)

The function of the switchyard is to transmit power from WCGS to the owner utility transmission system. The S4 system has no safety design basis.

The conductors from the main transformer high voltage terminals to the 345 kV switchyard bus are presently two (2) 954 ACSR (aluminum conductor steel reinforced) per phase. This installation limits the output to 2000 amperes per phase. The increase in unit output could potentially result in 2250 amperes per phase. Western Resources Engineering has evaluated the situation and determined that new conductor will be necessary to carry the additional amperes. The final design has not yet been finalized however it is anticipated that the new conductors will be 1590 ACSR. This will increase the line capacity to 3000 amperes.

Generator disconnect switch 345-55 is rated at 2000 amperes and is required to carry full unit output. The anticipated unit output could potentially be as much as 2250 amperes. Western Resources Engineering has determined that it is necessary to increase the capacity of the switch. The increase in the switch rating will be accomplished by a "line parts" change out which will increase the switch capacity to 3000 amperes.

The generator output breakers are rated at 2000 amps each continuous operation and are acceptable for the uprated conditions. However a concern would arise should the plant experience a breaker trip on either one of the two breakers, 345-50 or 345-60. In this case, all generator output current would be forced through one breaker which could exceed the rating of 2000 amps for the remaining breaker. Load reduction on the generator output would need to be performed to bring the breaker loading within the ampere rating of 2000 amps. The system is acceptable for operation at its present configuration, however the system is under evaluation to predict the conditions at which generator output would be reduced should either of the two breakers be lost.

H. Onsite Power and Electrical Distribution Systems

1. 4.16 KV AC Lower Medium Voltage (Class 1E) (NB)

The function of the lower medium voltage system is to receive power from the two 12/16 MVA Engineered Safety Features (ESF) transformers and distribute it to the two redundant load groups in the Class 1E system. Load Group 1 is served by 4.16 kV Bus NB01 and Load Group 2 is served by 4.16 kV Bus NB02.

The 4.16 kV Class 1E system distributes power to all safety-related loads, required for normal power generation, as well as for safe shutdown of the unit. Each 4.16 kV load group is supplied by two preferred power supply feeders and by one diesel generator supply feeder.

The ESF transformer is 3 phase, outdoor OA/FA, 12/16 MVA, 13.8 kV delta primary, 4.16 kV wye secondary with neutral grounded through resistor. The 4.16 kV switchgear is arranged in two buses, NB01 & NB02, and rated at 350 MVA, with a main bus continuous rating of 2,000 amperes and load breaker continuous rating of 1,200 amperes.

A review of the NB System protective relay settings indicates that no changes are required to support power rerate.

There are no changes required in ESF equipment loads due to power rerate. Based on the preceding, the existing Lower Medium Voltage System is adequate to support power rerate.

2. Standby Diesel Generator (Class 1E) (NE)

The function of the Standby Power Supply System is to provide power to the essential loads required for safe shutdown and isolation of the reactor in the event of a loss of the preferred power source.

The standby power supply system consists of two diesel generators, each connected exclusively to a single associated 4.16 kV Class 1E bus of a load group (Bus NB01 - Load Group 1; Bus NB02 - Load Group 2). Each diesel generator has a continuous rating greater than the sum of the loads required to satisfy the load demand caused by a LOCA and loss of the preferred power supply. Each diesel generator set capacity was established in accordance with Regulatory Guide 1.9, and each diesel generator is connected exclusively to a single associated 4.16 kV safety features bus with no provisions for parallel operation with the redundant diesel generator.

There are no changes required in ESF equipment loads due to power rerate. Therefore, the existing Standby Diesel Generator System is adequate to support power rerate.

3. 480 V AC Low Voltage (Class 1E) (NG)

The function of the Class 1E Low Voltage System is to distribute power at a nominal 480 volts for all low voltage safety-related loads. In addition, the Class 1E Low Voltage System furnishes power to certain selected loads which are not directly safety-related but are important to plant operation.

The NG system is divided into two load groups which are redundant and independent, Load Group 1 and Load Group 2. The system consists of Load Center Unit Substations and Motor Control Centers. The Load Center transformers are 1000 kVA, 3 phase, open ventilated dry type, 4.0 kV delta primary, 480 Volt wye secondary with grounded neutral. The bus rating of Motor Control Centers is 25 kA RMS symmetrical short circuit, with continuous current rating for each main and vertical bus of 600 amperes & 300 amperes respectively. No new 480 volt loads are added and the existing equipment loads are not revised due to power rerate. In addition, the NG System protective relay settings were reviewed to evaluate the impact of power rerating, and it was determined that no changes are required. Therefore, the existing Class 1E Low Voltage System is adequate to support power rerate.

4. 125V DC Power (Class 1E) (NK)

The function of the Class 1E 125 Volt DC system is to provide dc power to Class 1E dc loads and for control and switching of the Class 1E systems.

The Class 1E batteries are sized in excess of 150 percent of the system requirements to supply the required loads for 200 minutes. The capacity of each battery charger is based on the largest combined demand of all steady state loads and the recharging capacity required to restore the battery from the design minimum discharge state to the fully charged state within 12 hours.

In addition, the battery sizing was reviewed to evaluate the impact of power rerate, and it was determined that no changes are required.

No new Class 1E 125 Volt DC loads will be added, and the existing system loads will not increase due to power rerate. Therefore, the existing Class 1E 125 Volt DC Power System is adequate to support power rerate.

5. 120V Instrument AC Power (Class 1E) (NN)

The function of the Class 1E Instrument AC Power System supplies power for vital instrumentation and control loads required for safe shutdown of the reactor as well as power generation.

Four independent, Class 1E, 120 Volt vital instrument AC power supplies are provided to supply the four channels of protection and reactor control systems. Each vital instrument ac power supply consists of one inverter, one distribution bus, and one manual transfer switch. No new Class 1E, 120 Volt AC loads will be added, and the existing system loads will not be increased due to power rerate. Therefore, the existing NN System is adequate to support power rerate

6. 13.8 KV AC-Higher Medium Voltage (PA)

The function of the 13.8 kV Higher Medium Voltage System is to supply power at 13.8 kV for all non-safety related loads from each of the two 13.8 kV buses, PA01 and PA02. Each bus is supplied from one of the two secondary windings of both the start-up and the unit auxiliary transformers. The 13.8 kV system supplies power to the Reactor Coolant Pumps and on-site loads. The system also supplies power to the non-Class 1E 4.16 kV system (PB) and non-Class 1E 480 Volt System (PG). In addition, the 13.8 kV system also distributes power from one winding of the Startup Transformer to ESF Transformer XNBO2, thus providing one source of off-site power to the Class 1E 4.16 kV system (NB).

The 13.8 kV switchgear rating for the PA system was reviewed to determine the impact of the power rerate. The two 13.8 kV buses, PA01 & PA02, are rated at 1000 MVA, 15 kV with a main bus continuous current rating of 3000 ampere and are not adversely impacted by power rerate.

The 13.8 kV system supplies power to the four Reactor Coolant Pumps (RCP), two from each bus. Based on the power rerate parameters, the RCP motors are capable of operating continuously at the new hot loop load point which exceeds the nameplate rating of 7000 HP by 92 HP, and the new cold loop load of 907[illegible] HP, which represents an increase of 324 HP over the existing cold loop rating of 8750 HP. These increases in RCP motor loads have been reviewed by Westinghouse and found to be within the capabilities of the existing motors. The change in the motor HP has no impact on relay set points, breaker capacity or cable sizing.

Based on the preceding, the 13.8 kV Higher Medium Voltage System is adequate to support power rerate.

7. 4.16 KV AC-Lower Medium Voltage (PB)

The function of the 4.16 kV Lower Medium Voltage System is to receive power from the 13.8 kV system (PA) buses through two Station Service Transformers which step down the voltage to supply the two 4.16 kV buses, Bus PB03 and Bus PB04. The system provides power to non-safety-related 4 kV motors required to support normal power operation. The system includes two Station Service Transformers (XPB03 & XPB04) rated at 12/16 MVA, OA/FA, 13.8 kV delta primary, 4.16 kV wye secondary with neutral grounded through resistor. The two buses, PB03 and PB04, are rated at 350 MVA, 4.16 kV with a main bus continuous current rating of 2,000 amperes. The Condensate Pump motors (3,500 HP) and Heater Drain Pump motors (1,500 HP), fed from PB03 and PB04, will operate at slightly higher loads at power rerate, but will not exceed their nameplate ratings. Review of the ratings of the Station Service Transformers and 4.16 kV switchgear PB03 and PB04 indicates they are adequate to support power rerate. In addition, a review of the protective relay settings for the PB System indicates they are adequate to perform their protective functions at the power rerate loads. Based on the preceding, the PB System is adequate to support power rerate.

8. 480 VAC Low Voltage (non-Class 1E) (PG)

The function of the non-Class 1E Low Voltage System is to distribute power at a nominal 480 volts to all nonsafety-related loads. The PG System is divided into two groups which normally supply power to motors below 250 hp and other nonsafety-related loads required for the normal operation of the generating unit. The PG System has no safety design basis.

The load center unit substations consist of an incoming line section, transformer, and low voltage section with metal enclosed drawout power circuit breakers. The motor control center sections are divided into individual compartments for isolation of combination breaker, motor starter, and feeder tap load circuit breakers. The majority of the load center unit substations are supplied power from 13.8 KV buses PA01 and PA02.

No new loads will be added, and the existing system loads will not increase due to power rerate. Therefore, the existing PG System is adequate to support power rerate.

9. 250V DC (non-Class 1E) (PJ)

The function of the non-Class 1E 250 Volt DC System is to supply non-vital dc motors, such as the emergency lube oil pumps and emergency seal oil pumps. The PJ System has no safety design basis.

The rating of each PJ System battery is approximately 25 percent greater than that required to supply the total connected non-Class 1E loads required for power generation. The capacity of each battery charger is based on the largest non-continuous load (the running load of the largest motor for testing purposes), plus the capacity required to restore the battery to the fully charged state within 12 hours, and simultaneously feed all continuous loads required during normal plant operation.

In addition, the battery sizing was reviewed to evaluate the impact of power rerate, and it was determined that no changes are required.

No new loads will be added, and no existing loads are increased due to power rerate. Therefore, the existing 250 Volt DC System is adequate to support power rerate.

10. 125V DC (non-Class 1E) (PK)

The function of the non-Class 1E 125 Volt DC System is to supply non-safety related control and instrumentation loads, emergency lighting and provide an alternate source of power for the computer inverters. The system has no safety design basis.

The rating of each PK System battery is approximately 25 percent greater than that required to supply the total connected non-Class 1E loads required for power generation. The capacity of each battery charger is based on supplying the required loads, while simultaneously recharging the battery to the fully charged state within 12 hours from the design minimum charge state required during plant normal operation.

In addition, the battery sizing was reviewed to evaluate the impact of power rerate, and it was determined that no changes are required.

No new 125 Volt DC loads will be added, and no existing loads will increase due to power rerate. Therefore, the existing 125 Volt DC System is adequate to support power rerate.

11. Instrument AC Power (PN)

The function of the non-Vital Instrument AC power supply is to furnish reliable power to all nonsafety-related plant instruments. In addition, it is also utilized as the preferred source of power for the public address system. The PN system has no safety design basis. The Instrument AC power system is sized for full load operation. Power rerate will not add any additional equipment which require 120V AC power, nor increase the load on any existing PN System equipment. Therefore, the PN System is adequate to support power rerate.

I. BOP Computer and Process Sampling Systems

1. Balance of Plant Computer (RJ)

The primary function of the Balance of Plant (BOP) Computer is to aid the operators through data acquisition and alarm presentation. The BOP Computer also functions as a management information tool by reducing, storing, logging and transmitting data.

The BOP computer is not required for power generation, serves no safety function and is not essential for safe shutdown. An uninterruptible power supply, which is not impacted by the power rerate, provides power to the BOP computer system.

No new points have been identified to be added to the BOP system as a result of power rerate. A review of the Total Plant Setpoint Document (TPSD) and Reference 2 indicates no impact on BOP or NSSS alarm setpoints. Based on the preceding, the BOP Computer is adequate to support power rerate.

2. Process Liquid Sampling and Analysis (RM)

The function of the Process Liquid Sampling system is to provide representative samples of non-nuclear process fluids to facilitate analysis of fluid properties necessary for plant operation, corrosion control and monitoring of equipment and system performance. The process sampling system serves no safety function and is not required for safe shutdown. The process sampling system continuously monitors the water quality of secondary plant systems.

Balance of plant process conditions will not change significantly due to power rerate. The RM System's sample conditioning equipment is capable of handling the changes to system process conditions, therefore, the Process Liquid Sampling and Analysis System is adequate to support power rerate.

3. Nuclear Sampling (SJ)

The function of the Nuclear Sampling System and the Post Accident Sampling System (PASS) are to provide representative samples of process fluids to facilitate radiological and chemical analysis necessary for plant operation, corrosion control and monitoring of equipment and system performance during normal operation and post-accident. Except for piping and valves associated with containment penetrations, which are designed to ASME Section III Class 2 and meet Seismic Category I requirements, the Nuclear Sampling System and PASS perform no safety function and are not essential to safe shutdown.

The Nuclear Sampling System, consisting of the primary (reactor coolant and auxiliary systems) and radwaste sampling systems, is an intermittent, manually operated system with the exceptions of continuous monitoring of the steam generator blowdown samples for radioactivity and reactor coolant letdown fluid sampling for failed fuel monitoring. Both on-line and grab sampling capabilities are available from the Post-Accident Sampling System. Primary system process conditions will not change significantly due to power rerate, nor will postulated post-accident process fluids. The Nuclear Sampling System and Post Accident Sampling System sample conditioning equipment is capable of handling the minor changes in Primary and Post-Accident process fluids due to power rerate. Therefore, the SJ System is adequate to support power rerate.

V WCGS PROGRAM REVIEWS

A. Pressurizer Sub-compartment Pressure/Temperature Analysis

The mass and energy releases from the pressurizer surge line break are currently bounding for the pressurizer sub-compartment. Westinghouse indicates that the mass and energy releases from the surge line break are strongly affected by the initial temperature conditions of the fluid (RCS), and that due to the reduction in the initial temperature (Case 3), the peak mass and energy releases will increase by 15% due to power rerate.

In order to evaluate the effect of this increase on the pressurizer sub-compartment, this sub-compartment pressure-temperature analysis was rerun using a uniform 15% increase in all time dependent mass/energy release values. The results of this re-analysis indicate an increase in the peak pressures of up to 2.5 psi, and an increase in peak temperatures of up to 6^{O}F. However, the new pressure values remain below the original compartment structural design pressure. The new temperature values have no impact on existing analyses, and remain below the temperatures used for qualification of equipment in containment. Table V-1 provides a comparison of the original values and the new values applicable for the limiting power rerate case (Case 3).

Table V-1

Comparison of Pressurizer Sub-Compartment Peak Pressure/Temperature Response to a Pressurizer Surge Line Break

	Existing Results					Power Rerate Results				
COMPARTMENT	P_{max} (psia)	P_{max} (psig)	Time (sec)	T_{max} (F)	Time (sec)	P_{max} (psia)	P_{max} (psig)	Time (sec)	T_{max} (F)	Time (sec)
1	22.1	7.3	.039	226.0	.156	23.6	8.9	.056	231.4	.147
2	20.5	5.8	.042	221.3	.185	21.4	6.7	.041	225.6	.180
3	15.4	0.7	.505	212.6	.505	15.6	0.9	.500	213.9	.500
4	26.7	12.0	.014	237.9	.065	28.3	13.6	.014	243.3	.080
5	19.0	4.3	.069	223.9	.124	20.4	5.7	.055	228.0	.119
6	26.9	12.2	.015	240.1	.059	29.4	14.7	.016	245.9	.044
7	24.9	10.2	.028	229.1	.164	26.7	12.0	.028	235.1	.160
8	22.8	8.1	.043	223.1	.352	24.4	9.7	.067	226.7	.280
9	21.7	7.0	.057	219.2	.505	22.6	7.9	.055	221.3	.500
10	15.6	0.9	.505	212.8	.505	15.8	1.1	.500	214.5	.500
11	15.3	---	.505	120.0	.016	15.4	0.7	.500	120.0	.015

Note: The above results are based on initial conditions of 120F, 14.7 psia, and 50% relative humidity in all sub-compartments.

Based on the results of the Pressurizer Sub-compartment Pressure/Temperature re-analysis, it is concluded that power rerate will not have an adverse impact on the pressurizer sub-compartment.

B. Radiological Source Terms for Environmental Qualification and Shielding Design

Radiological doses due to power uprating would normally be expected to increase in proportion to the change in reactor core power, or approximately 4.5% for the proposed WCGS power rerate. Therefore, a review was performed to determine the impact of power rerating on WCGS shielding and equipment qualification (EQ) radiological doses, inclusive of the change in fuel cycle from 12 months to 18 months, to support the power rerate program. The results of this review yield the following increases in radiological doses:

1. ~8% increase in doses from Source A

2. ~42% increase in doses from Sources B and C

3. < 1% increase in airborne gamma doses in containment

Where: 1. Source A = Radioiodines and Noble Gas source
2. Source B = Reactor Coolant System source post-LOCA
3. Source C = Containment Sump source post-LOCA

1. The estimated 8% increase in Source A doses result in less than a 1% increase in total gamma energy, and an approximate 8% increase in the semi-infinite beta dose in containment. These increases will have no effect on shielding design, as the existing margins in the WCGS EQ and shielding designs far exceed the impact of these increases. The effect of these dose increases is not expected to impact equipment qualification.

2. The estimated 42% increase in doses from Sources B and C is due primarily to Cesium 134. The NUREG-0588 design basis for the WCGS conservatively assumed 50% of the total available Cesium is released to the containment at time zero, post-LOCA. Due to conservatisms in the original equipment specifications, most components were originally qualified to the resultant radiation levels. For the isolated cases where the 50% Cesium source term proved too severe, the equipment was evaluated against, and qualified to the 1% Cesium source term.

For a limited number of the components originally qualified to the 50% Cesium source term, the new power rerate doses may be too severe. In such cases, the affected components will be reviewed to confirm they are qualified for the 1% Cesium source term. For components originally qualified to the 1% Cesium source term, the estimated increase in dose is approximately 5%, and therefore, are not expected to be limiting for EQ due to the existing design margins.

3. The estimated increase of less than 1% in the airborne gamma doses in containment is not expected to be limiting for EQ or shielding due to the existing design margins.

Based on the preceding, it is concluded that radiological doses as applied to plant shielding and environmental qualification of safety-related electrical equipment will not be limiting for power rerate.

C. ALARA

Radiation sources used as the basis for the design of personnel protection, including ALARA, were based on a reactor thermal power level of 3565 MWt, which is equivalent to the proposed power rerate reactor core power level. ALARA reviews were begun during the initial design phase of WCGS, and continued through final design and construction. Design considerations and features intended to ensure ALARA doses included (1) the incorporation of special design considerations into equipment specifications, (2) application of the (design) Scale-Model Program, (3) ALARA design reviews, and, (4) the implementation of radiation protection design features such as the use of a packless, low-leakage, ball-type pressurizer spray valve, provisions for the continuous stripping of noble gases from the reactor coolant system, the provision of a mini-purge system to permit purging of the containment during power operation, prior to operator access, and the provision of a containment atmospheric control system to remove air-borne iodine from the containment.

The crud levels in the reactor coolant are controlled by reactor coolant water chemistry control, corrosion rates and materials. These aspects will not change due to power rerate, therefore, there will not be a significant change in crud levels. From this, and the design features incorporated into the original plant design, it can be concluded that power rerate will not result in a significant impact on the WCGS ALARA program.

D. Hazards

1. Piping Dynamic Loads

In order to assess the impact of power rerate on piping dynamic loads due to transients (e.g., turbine trip, relief valve lifts, main feedpump trip), the bases of the piping hydraulic calculations for the following systems were reviewed against the power rerate program process parameters:

AB	-	Main Steam
AC	-	Main Turbine
AE	-	Main Feedwater
BM	-	Steam Generator Blowdown
DA	-	Circulating Water
FB	-	Auxiliary Steam
FC	-	Auxiliary Turbines
GT	-	Containment Purge

Based on the results of this review, it was determined that, in most cases, the original analyses bound the power rerate program process parameters, due to conservatisms incorporated into the original analyses. However, in the case of the Main Steam, Main Turbine, Main Feedwater, and Containment Purge Systems, some degree of increase in forces were identified for certain transients, ranging from 2% to 17%. In each of these cases, the increase in dynamic loads was evaluated against the piping stress and pipe support analyses to confirm piping stresses remain within allowables, and support loads remain within acceptable design limits. Based on the results of these evaluations, all system piping and supports remain within allowable stresses and design loads for power rerate. Therefore, it can be concluded that the dynamic loads which result from operational transients (e.g., relief valve lifts, main feedpump trip, turbine trip) at the power rerate conditions will not adversely impact the above systems.

2. Pipe Break

In order to assess the impact of changes in process parameters due to power rerate on high energy line break (HELB) analyses for WCGS, the bases of the WCGS pipe break calculations for the following high energy systems were reviewed against the power rerate program process parameters:

AB - Main Steam
AE - Main Feedwater
BB - Reactor Coolant
BG - Chemical and Volume Control
BM Steam Generator Blowdown
EM High Pressure Coolant Injection
EP - Accumulator Safety Injection
FB - Auxiliary Steam
FC - Auxiliary Turbines
HB - Liquid Radwaste

Based on the results of this review, the forcing functions for Pipe Whip and Jet Impingement due to postulated pipe breaks were originally developed using methodology based on one of the following three (3) conditions:

1. Superheated or Saturated Steam Break Analysis

2. Saturated or Subcooled Water Break Analysis

3. Cold Water Break Analysis

In all cases, the original analyses are based on the maximum upset operating process conditions, or were performed assuming (conservatively) cold water. These conditions bound all normal operating and accident power rerate parameters, therefore, the existing pipe break analyses bound for power rerate.

3. Flooding

The flooding rates estimated in the original WCGS flooding analyses are based on the worst-case pipe failures in each room containing safety-related equipment. Calculated flood levels are based on automatic isolation or operator action after a reasonable time delay following detection of flow from the applicable pipe cracks or breaks. The operator action time is 30 minutes plus any time required for the operator to travel to a location outside of the main control room. For power rerate, flow rates in the main steam, main feedwater and condensate systems will increase, as compared to the current 100% power conditions. The safety-related plant areas which could be impacted by flooding due to an increase in the Main Feedwater, Main Steam and Condensate System flow rates, are the Main Steam/Main Feedwater Isolation Valve (MSIV/MFIV) Compartment and the Containment.

The maximum flood level in the MSIV/MFIV Compartment was originally based on a main feedwater line break. The flood level in this area is based on the storage capacity of the condenser hotwell, assuming the condensate pumps trip on low condenser hotwell level. Since neither the condenser hotwell storage capacity, nor the condenser hotwell low level setpoint require revision for power rerate, the postulated flooding in this area will not be impacted by power rerate.

The maximum calculated flood level inside primary containment is elevation 2004'-6", due to a LOCA. The next most limiting flood level inside primary containment is elevation 2004'-5", and is the result of a Main Steam Line Break (MSLB). This level was calculated based on a conservative operator action time (31.5 minutes), and mass released into the containment from several sources, including Main Steam blowdown, Auxiliary Feedwater addition and the Reactor Water Storage Tank. The total calculated mass addition from all sources is approximately 3.8×10^6 lbm. The total mass released due to Main Steam blowdown, which is the only source that will increase due to power rerate, is less than 8% of the total. Therefore, no significant change in the maximum flood level due to power rerate is expected, since the main steam flow increases by less than 1%.

VI CONCLUSIONS

A detailed review for the impact of power rerating from a reactor power level of 3411 MWt to 3565 MWt has been conducted. In addition, review of WCGS Programs to evaluate the impact of power rerate on 1) the consequences of a Pressurizer Surge Line Break, 2) radiological source terms used for Environmental Qualification of equipment and plant shielding design, 3) the WCGS ALARA Program, 4) piping dynamic loads due to transients, and 5) the consequences of High Energy Line Breaks have also been conducted.

Based on the review results presented in this report, the WCGS BOP and supporting systems will support a power rerate from the currently licensed reactor power level of 3411 MWt to an uprated reactor power level of 3565 MWt. Further, the WCGS Programs will not be adversely impacted by power rerate. This power rerate is estimated to yield an increase of 28 MW of turbine-generator electrical power output, based on the target operating RCS T_{Hot} reduction of 5^oF. Due to flow limitations in the main turbine, operation at 104.5% of the current licensed reactor power, at the Case 3 power rerate conditions would require hardware modifications to the main turbine, which are not planned at this time.

3.7 OTHER SYSTEMS

3.7.1 Spent Fuel Pool

The current licensing basis thermal-hydraulic analysis for the Spent Fuel Pool (SFP) storage facility and accessories was performed with a heat load based on the decay heat generation from the spent fuel assemblies (FAs) stored in the pool. The decay heat generated by these spent fuel assemblies were calculated based on the assumptions that they have undergone different irradiations in the reactor at an uprated reactor power of 3565 MWt.

The results of the SFP thermal hydraulic analysis show that the SFP cooling system is capable of maintaining the SFP water temperature under the design limit of 135^{o}F, with one of the two fuel pool cooling trains operating and a heat load based on the decay heat generation from a number of fuel assemblies that are projected to be discharged from the core and placed in the pool 175 hours after reactor shutdown, plus 19 previous refueling batches already stored in the pool. With one fuel pool cooling train operating, the analysis result also indicates that the pool temperature is maintained at or below 160^{o}F, assuming the full core offload has occurred 196 hours after reactor shutdown and the spent fuel storage racks are full (maximum storage condition). This calculated "full core offload" bulk temperature of the pool water is well below the boiling temperature.

Based on these analysis results, it is concluded that the existing SFP cooling system has adequate cooling capacity to accommodate the additional spent fuel heat load as a result of implementing design change to operate the reactor at a higher power level and, therefore, any conclusions that were based on the current licensing basis SFP thermal hydraulic analysis remain valid for the power rerate program.

3.7.2 Steamline Break Mass/Energy Release Outside Containment

Steamline ruptures occurring outside the reactor containment structure may result in significant releases of high energy fluid (superheated steam) to the equipment surrounding the steam systems. The impact of the steam releases on this equipment depends upon the mass flowrate and enthalpy of the steam which is determined by the plant configuration at the time of the break, the plant response to the break, as well as the size and location of the break. Because of the interrelationship between many of the factors which influence steamline break mass and energy releases, it is practically impossible to determine a single "worst" case with respect to mass and energy release. As a result, the steamline break mass/energy release calculations are analyzed for a range of conditions. Specifically, a spectrum of steam line break sizes (4.6 ft^2, 1.0 ft^2, 0.7 ft^2 and 0.5 ft^2) at different power levels (102% and 70%) in the main steam tunnel were postulated for the SNUPPS utilities. A break size of less than 0.5 ft^2 was not considered because it did not result in significant superheating of the steam until after operator response time for terminating auxiliary feedwater.

In 1984, Wolf Creek joined the High Energy Line Break/Superheated Blowdowns Outside Containment Subgroup of the Westinghouse Owners Group to sponsor a program that addressed NRC concerns on the effect of superheated steam releases on the environmental qualification of equipment located outside containment. Steamline break mass and energy release analyses were performed to provide the participating plants with data which could be used in plant-specific equipment qualification evaluations. The results from these analyses were documented in WCAP-10961, "Steamline Break Mass/Energy Releases for Equipment Environmental Qualification Outside Containment". For the analyses documented in WCAP-10961, the WCGS was included in a plant category (Category 1) based on plant size and power level (4 loop, 3425 MWt), steam generator type (D4), and steamline break protection system design. The WCGS conform to these parameters with the exception of steam generator type since the WCGS has Model F steam generators. A sensitivity study was performed which compared Model F, Model D4, Model D5 and Model 51 steam generator using actual steam generator operating characteristics. This study demonstrated that use of the Model D4 steam generator was conservative for the WCGS in that peak break enthalpy is 13 Btu/lb lower for the Model F steam generator for the steamline break mass/energy release results.

Sensitivity studies on core power have indicated that there is a small increase in break flow, break energy and superheat enthalpy as a result of increased power level, e.g., 1 Btu/lb in peak enthalpy between 3425 MWt and 3579 MWt. The use of the Model D4 steam generator in the analysis ensures a conservative mass/energy release even for a power level of 3579 MWt, because the peak break enthalpy is conservative by approximately 12 Btu/lb and the integrated break flow is lower by approximately 5% for the Model F steam generators.

Based upon the above information, the calculated mass/energy release data that were used in the previous evaluations of environmental qualification of equipment remain bounding for the WCGS power uprating program.

4.0 DOCUMENT REVISIONS

4.1 EMERGENCY OPERATING PROCEDURES (EOP) ASSESSMENT

EOP setpoint calculations and the corresponding EOP page changes have been generated for the Wolf Creek rerating program. These changes have been based on an NSSS power level of 3579 MWth, a reduced thermal design flow of 93,600 gpm per loop, 10% average/10% peak steam generator tube plugging, increased core bypass flow to 8.4%, and a T_{AVG} operating range from 570.7°F to 588.4°F. Implementation of the following EOP setpoint calculations and revisions will be conducted concurrent with plant operation at the uprated power level.

Reactor Coolant Pump (RCP) Trip Pressure
RCP Trip Pressure, Adverse Containment
Safety Injection Flow vs. Decay Heat During Loss of Emergency Coolant Recirculation
Saturation Pressure for 100% Power T-HOT
Saturation Pressure for 100% power T-HOT, Adverse Containment
Steam Generator Pressure Limit to Prevent Nitrogen Injection into the RCS
T-HOT Limit to Prevent Nitrogen Injection into the RCS

4.2 SUMMARY OF TECHNICAL SPECIFICATION IMPACT

In accordance with 10CFR50.90 regarding application for amendment of an operating license, changes to the Technical Specifications resulting from the rerating program have been defined and have been submitted to the NRC for review. Those changes to the Technical Specifications are discussed in Section 5.0, and are supported by an accompanying significant hazard evaluation as required by 10CFR50.92 and an environmental impact determination as required per 10CFR51.21.

Table 4-1
Summary of Technical Specification Changes for Power Uprate

Page	Section	Description of Change	Justification
1-5	1.25	Rated Thermal Power	Definition of Core Rated Thermal Power changed from 3411 MWt to 3565 MWt
2-4	Table 2.2-1	Revised TA, Z, and S terms	OTΔT / OPΔT changes due to revised ΔT
3/4 2-15	Table 3.2-1	Revised maximum indicated RCS T_{AVG}	Reduced operating temperature
B 3/4 2-6	3/4.2.5	Revised maximum indicated RCS T_{AVG}	Reduced operating temperature

4.3 SUMMARY OF USAR ASSESSMENT

Paragraph (e) of 10CFR50.71 provides the requirement to periodically update the contents of the USAR originally submitted as part of the application for the operating license. The intent is to maintain information in the USAR as the latest material which actually describes the current plant configuration. The information in the update is to include the effects of any changes made to the facility or procedures as described in the USAR. In compliance with this regulation, revised sections of the Wolf Creek USAR have been developed as appropriate which reflect the analyses and evaluations that take into account operation at the rerated conditions. These revisions will be incorporated into the Wolf Creek USAR on a schedule consistent with the annual USAR update program.

5.0 NO SIGNIFICANT HAZARDS EVALUATION (10CFR50.92)

Pursuant to 10CFR50.92, each application for amendment to an operating license must be reviewed to determine if the proposed change involves a significant hazards consideration. The proposed Technical Specification amendment, relative to operation of Wolf Creek Generating Station (WCGS) at an uprated power level, has been reviewed and deemed not to involve significant hazards considerations. The basis of this determination is presented below.

Background

Wolf Creek Nuclear Operating Corporation (WCNOC) has completed a long term program of analyses and evaluations designed to demonstrate that WCGS can safely operate at a core power level of 3565 MWth. Definition 1.25 of the WCGS Technical Specifications currently limits the maximum power output of the reactor core to 3411 MWth. This value is defined as RATED THERMAL POWER (RTP) and is utilized extensively throughout the Technical Specifications to define applicability as a limit value, action statement condition, or surveillance requirement. The primary goal of the rerating program involves changing the definition of RTP to 3565 MWth, which will allow operation of the reactor at the higher power output.

Analysis

The NSSS systems and components and the BOP were reviewed to determine the impact of rerating. The review verified that the safety, functional, and structural criteria as defined in the USAR are met using the rerated conditions. The review included the aspects of design and operation that are potentially affected by operating at rerated conditions. It was performed in accordance with the licensing criteria and standards that currently apply to WCGS. Equipment design was evaluated against the current design requirements. Evaluations and analyses were performed as appropriate to definitively determine that WCGS is capable of safe reliable operation at rerated conditions without any physical modifications.

A thorough review of the accident analyses in the WCGS USAR has been performed to determine those events sensitive to the increase in RTP. As a result, each of those events so identified have been reanalyzed to determine that the various acceptance criteria are still met, assuming an increase in reactor core power. T-HOT reduction, steam generator tube plugging, reduced thermal design flow and peaking factor increase assumptions were also made to generate operating flexibility. In all cases, the acceptance criteria for all transients were maintained and therefore the margin of safety is maintained.

Results

Based on the information presented above and by reference, the following conclusions can be reached with respect to 10CFR50.92 for the increase in RTP and change in operating T_{HOT}.

1. The increase in RTP and change in operating T_{HOT} does not involve a significant increase in the probability or consequences of an accident previously evaluated. Operation at this power level will not cause any design or analysis acceptance criteria to be exceeded. The structural and functional integrity of the plant systems is maintained. Rated thermal power is an input assumption to the equipment design and accident analyses but it is not itself an initiator for any transient. Therefore, the probability of occurrence is not affected.

   The increase in RTP and change in operating T_{HOT} does not affect the integrity of the fission product barriers utilized for mitigation of radiological dose consequences as a result of an accident. The offsite dose predictions remain within the acceptance criteria for each of the transients affected. Since it has been determined that the limiting transient results are unaffected by the power increase and T_{HOT} reduction, it is concluded that the consequences of an accident previously evaluated are not increased.

2. The increase in RTP and change in operating T_{HOT} does not create the possibility of a new or different kind of accident from any accident previously evaluated because no new operating configuration is being imposed that would create a new failure scenario. In addition, no new failure modes are being created for any plant equipment. System and component design bases have been reviewed to determine that the different cyclic temperature transients resulting from rerating do not significantly affect the fatigue life of the equipment. Therefore, the types of accidents defined in the USAR continue to represent the credible spectrum of events to be analyzed which determine safe plant operation.

3. The increase in RTP and change in operating T_{HOT} does not involve a significant reduction in a margin of safety because RTP and operating temperature are two of the inherent assumptions that determines the safe operating range defined by the accident analyses, which are in turn protected by the Technical Specifications. The acceptance criteria for the accident analyses are conservative with respect to the operating conditions defined by the Technical Specifications. The work performed for the rerating program confirms that the accident analyses criteria are met at the revised configuration. Therefore, the adequacy of the revised Technical Specifications to maintain the plant in a safe operating range is also confirmed and the increase in RTP and change in operating T_{HOT} does not involve a significant reduction in a margin of safety.

Conclusion

Based upon the preceding analysis, it has been determined that the proposed change to the Technical Specifications to increase the definition of RTP and change in operating T_{HOT} does not involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated or involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed change meets the requirements of 10CFR50.92(c) and does not involve a significant hazards consideration.

6.0 ENVIRONMENTAL IMPACT DETERMINATION

This amendment request meets the criteria specified in 10 CFR 51.22(c)(9). Specific criteria contained in this section are discussed below.

(i) this amendment involves no significant hazards consideration

As demonstrated in Section 5, this proposed amendment does not involve any significant hazards considerations.

(ii) there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

This proposed license and technical specification change involves increasing reactor thermal power approximately 4.5% and allowing operation at reduced T_{HOT} temperatures (as much as 15 degrees F below current values). These types of operational changes will not affect the types of effluents since no new process is involved. However, some increase in radiological doses from the reactor coolant system source terms inside the containment will occur. These increases will not affect the shielding as the existing margins for exceed the impact of the increases. In addition, the increases would be limited to the containment and would not be expected to significantly impact the offsite release of effluents. Therefore, the revised source terms will not result in any significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

(iii) there is no significant increase in individual or cumulative occupational radiation exposure

As described above, the analysis shows the revised source term changes will not affect the shielding design margins and are limited to areas within containment. Therefore, there will be no significant increase in individual or cumulative occupational radiation exposure associated with the proposed change.

Based on the above, there will be no significant impact on the environment resulting from this change and the change meets the criteria specified in 10 CFR 51.22(c)(9) for a categorical exclusion from the requirements of 10 CFR 51.21 relative to a specific environmental assessment by the Commission.