ENCLOSURE 1
FirstEnergy Nuclear Operating Company
Beaver Valley Power Station, Unit Nos. 1 and 2

License Amendment Request Nos. 289 and 161
INCREASE CORE RATED THERMAL POWER BY 1.4% TO 2689 MWt

Attached is the Beaver Valley Units 1 and 2, 1.4-Percent Power Uprate Program, FENOC Licensing Submittal, January 2001.

FENOC

FirstEnergy Nuclear Operating Company

Beaver Valley Units 1 and 2, 1.4-Percent Power Uprate Program, FENOC Licensing Submittal

January 2001

TABLE OF CONTENTS

LIST OF ACRONYMS

AC	Alternating Current
AFWS	Auxiliary Feedwater System
ANSI	American National Standards Institute
ART	Adjusted Reference Temperature
ASDV	Atmospheric Steam Dump Valve
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
AVB	Anti-Vibration Bar
BHP	Brake Horsepower
B&PV	Boiler and Pressure Vessel
BOP	Balance Of Plant
BOL	Beginning Of Life
C&FS	Condensate and Feedwater System
CCW	Component Cooling Water
CFR	Code of Federal Regulations
CLH	Capped Latch Housing
COMS	Cold Overpressure Mitigation System
CRDM	Control Rod Drive Mechanism
CVCS	Chemical and Volume Control System
CWS	Circulating Water System
DC	Direct Current
DNB	Departure from Nucleate Boiling
E/C	Erosion/Corrosion
ECCS	Emergency Core Cooling System
EFPY	Effective Full-Power Year
EOL	End Of Life
EPRI	Electric Power Research Institute
ERG	Emergency Response Guideline
ESDR	Engineered Safeguards Design Rated
ESF	Engineered Safety Feature
ESFAS	Engineered Safety Feature Actuation System
FENOC	FirstEnergy Nuclear Operating Company
FER	Final Environmental Report
FR	Federal Register
GL	Generic Letter
HDS	Heater Drain System
ID	Inner Diameter
IFM	Intermediate Flow Mixer

LIST OF ACRONYMS (cont.)

LAR	License Amendment Report
LBB	Leak Before Break
LBLOCA	Large-Break Loss-Of-Coolant Accident
LEFM	Leading Edge Flow Meter
LOCA	Loss-Of-Coolant Accident
LOOP	Loss of Offsite Power
LSIV	Loop Stop Isolation Valve
LTCC	Long-Term Core Cooling
LWS	Laser Weld Sleeve
$\dot{M}_{steam}$	Mass Flowrate (Steam)
MFCV	Main Feedwater Control Valve
MFIV	Main Feedwater Isolation Valve
MPT	Main Power Transformer
MSIV	Main Steam Isolation Valve
MSNV/TV	Main Steam Non-return Valve/Trip Valve
MSS	Main Steam System
MSSV	Main Steam Safety Valve
MSVR	Main Steam Valve Room
Mva	Megavolt-ampere
MWe	Megawatt Electric
NPDES	National Pollutant Discharge Elimination System
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OBE	Operating Basis Earthquake
OD	Outer Diameter
OPΔT	Over-Power Delta-T
OTΔT	Over-Temperature Delta-T
P_{steam}	Steam Pressure
P&I	Proportional and Integral
PORV	Power-Operated Relief Valve
PPDWST	Primary Plant Demineralized Water Storage Tank
PTS	Pressurized Thermal Shock
PWR	Pressurized Water Reactor
RCCA	Rod Cluster Control Assembly
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RHRCV	Residual Heat Release Control Valve
RHRS	Residual Heat Removal System

LIST OF ACRONYMS (cont.)

RTD	Resistance Temperature Detector
RTDP	Revised Thermal Design Procedure
RTP	Rated Thermal Power
RT_{PTS}	Pressurized Thermal Shock
RTS	Reactor Trip System
RWST	Refueling Water Storage Tank
SBLOCA	Small-Break Loss-Of-Coolant Accident
SBO	Station Blackout
SD	Heater Drain System
SG	Steam Generator
SGTP	Steam Generator Tube Plugging
SGTR	Steam Generator Tube Rupture
SI	Safety Injection
SIS	Safety Injection System
SLB	Steam Line Break
SSE	Safe Shutdown Earthquake
SSST	System Station Service Transformer
T_{avg}	Vessel Average Temperature
T_{cold}	Vessel/Core Inlet Temperature
T_{hot}	Vessel Outlet Temperature
T_{steam}	Steam Temperature
TDF	Thermal Design Flow
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
USE	Upper Shelf Energy
USST	Unit Station Service Transformer
V&V	Verification and Validation
VCT	Volume Control Tank
V5H	Vantage 5 Hybrid

1.0 BACKGROUND AND REASON FOR THE PROPOSED CHANGE

Beaver Valley Units 1 and 2 are presently licensed for a core thermal power rating of 2,652 MWt. Through the use of more accurate feedwater flow measurement equipment, approval is sought to increase this core power by 1.4 percent to 2,689 MWt. The impact of a 1.4-percent core power uprate for applicable systems, components, and safety analyses has been evaluated.

This FirstEnergy Nuclear Operating Company (FENOC) 1.4-percent core power uprate for Beaver Valley Units 1 and 2 is based on eliminating unnecessary analytical margin originally required of emergency core cooling system (ECCS) evaluation models performed in accordance with the requirements set forth in the Code of Federal Regulations (CFR) 10CFR50, Appendix K (Emergency Core Cooling System Evaluation Models, ECCS).

The Nuclear Regulatory Commission (NRC) recently approved a change to the requirements of 10CFR50, Appendix K (as revised by the Federal Register (FR) 65 FR 34913, June 1, 2000). The change provides licensees with the option of maintaining the 2-percent power margin between the licensed power level and the assumed power level for the ECCS evaluation, or applying a reduced margin for ECCS evaluation. For the reduced margin for ECCS evaluation case, the proposed alternative reduced margin has been demonstrated to account for uncertainties due to power level instrumentation error. Based on the proposed use of the Caldon Leading Edge Flow Meter (LEFM) instrumentation to determine core power level with a power measurement uncertainty of less than 0.6 percent, it is proposed to reduce the licensed power uncertainty required by 10CFR50, Appendix K, for modest increases of up to 1.4 percent in the license power level using current NRC-approved methodologies.

The basis for the amendment request is that the Caldon instrumentation provides a more accurate indication of feedwater flow (and correspondingly reactor thermal power) than assumed during the development of Appendix K requirements. Complete technical support for this conclusion is discussed in detail in Caldon Topical Report ET-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM System," as approved in NRC's Safety Evaluation for TU Electric, dated March 8, 1999, and supplemented by Caldon Engineering Report ER-157P, Revision 2, December 2000, provided in Enclosure 2 of this letter. The improved thermal power measurement accuracy obviates the need for the full 2-percent power margin assumed in Appendix K, thereby increasing the thermal power available for electrical generation. It should be noted that amendments, LAR-286 (Unit 1) and LAR-158 (Unit 2), have been submitted that implement the use of Revised Thermal Design Procedure (RTDP) methodology. This methodology generates departure from nucleate boiling (DNB) margin, which will support the 1.4-percent uprating using the LEFM systems.

The desired power increase of 1.4 percent will be accomplished by increasing the electrical demand on the turbine generator. As a result of this demand increase, steam flow will increase and the resultant steam pressure will decrease. The reactor coolant system (RCS) average temperature will be maintained. The RCS hot leg temperature will increase and the cold leg temperature will decrease in response to the increased steam flow demand.

New procedures for maintenance and calibration of the LEFM system will be developed per the design control process based on the vendor's recommendations. Should the LEFM system be unavailable at one of the Beaver Valley units, the feedwater flow venturis will be used to sense feedwater flow rate, as was done prior to the installation of the LEFM. The core power reduction will be based on the original licensed power level and further supported by the RTDP WCAP-15264 Revision 3 for Unit 1 and WCAP-15265 Revision 2 for Unit 2. These reports provide the basis for the RTDP uncertainties that are used in the Beaver Valley safety analyses.

2.0 DESCRIPTION OF THE PROPOSED CHANGE

The proposed license amendment would revise the Beaver Valley Units 1 and 2 operating licenses and Technical Specifications (TSs) to increase the core power level by 1.4 percent to 2689 MWt. The power uprate is based on the use of the Caldon Leading Edge Flow Meter (LEFM) for determination of main feedwater flow and the associated determination of reactor power through the performance of the power calorimetric, currently required by Beaver Valley TSs. Specifically, the proposed changes are provided by the markups of the current Beaver Valley Units 1 and 2 operating licenses and TSs, in Attachments A-1 and A-2 of FENOC License Amendments Requests 289 and 161.

3.0 SAFETY ANALYSIS

3.1 APPROACH

The 1.4-Percent Power Uprate Program for Beaver Valley Units 1 and 2 has been completed consistent with the methodology established in WCAP-10263, "A Review Plan for Uprating the Licensed Power of a PWR Power Plant," issued in 1983. Since its submittal to the Nuclear Regulatory Commission (NRC), the methodology has been successfully used as the basis for power uprate projects on over 20 pressurized water reactor (PWR) units, including Diablo Canyon Units 1 and 2, Turkey Point Units 3 and 4, and Comanche Peak Unit 2.

The methodology in WCAP-10263 establishes the general approach and criteria for uprate projects including the broad categories that must be addressed, such as nuclear steam supply system (NSSS) performance parameters, design transients, systems, components, accidents, and nuclear fuel as well as interfaces between the NSSS and balance-of-plant (BOP) systems. Inherent in this methodology are key points that promote correctness, consistency, and licensability. No new analytical techniques have been used to support the 1.4-percent power uprate project. The key points include the use of:

- Well-defined analysis input assumptions/parameter values
- Currently approved analytical techniques
- Applicable licensing criteria and standards

The evaluations and analyses described herein have been completed consistent with this methodology for the Beaver Valley Units to increase licensed core power from 2,652 MWt to 2,689 MWt. Section 3.3 of this report discusses the revised NSSS design thermal and hydraulic parameters that were modified as a result of the 1.4-percent uprate and that serve as the basis for all of the NSSS analyses and evaluations. Section 3.4 concludes that no design transient modifications are required to accommodate the revised NSSS design conditions. Sections 3.5 through 3.7 present the systems (e.g., safety injection, residual heat removal (RHR), and control systems) and components (e.g., reactor vessel, pressurizer, reactor coolant pumps (RCPs), steam generator, and NSSS auxiliary equipment) evaluations completed for the revised design conditions. Section 3.8 summarizes the effects of the uprate on the BOP (secondary) systems based upon a heat balance evaluation. Section 3.9 provides an analysis of the effects of the power uprate on the Beaver Valley electrical power systems. Section 3.10 provides the results of the accident analyses and evaluations performed for the steam generator tube rupture, loss-of-coolant-accident (LOCA), and non-LOCA areas. Sections 3.11 and 3.12 summarize the containment accident analyses and evaluations and the radiological consequence evaluations. Section 3.13 contains the results of the fuel-related analyses. The results of all of the analyses and evaluations performed demonstrate that all acceptance criteria continue to be met.

3.1.1 General Licensing Approach for Plant Analyses Using Plant Power Level

The reactor core and/or NSSS thermal power are used as inputs to most plant safety, component, and system analyses. These analyses generally model the core and/or NSSS thermal power in one of four ways.

First, some analyses apply an explicit 2-percent increase to the initial condition power level to account solely for the power measurement uncertainty. These analyses have not been re-performed for the 1.4-percent uprate conditions because the sum of increased core power level (1.4 percent) and the decreased power measurement uncertainty (less than 0.6 percent) falls within the previously analyzed conditions.

The power calorimetric uncertainty calculation described in Section 3.10.5.1 indicates that with the Leading Edge Flow Meter (LEFM) devices installed, the power measurement uncertainty (based on a 95-percent probability at a 95-percent confidence interval) is less than 0.6 percent. Therefore, these analyses only need to reflect a 0.6-percent power measurement uncertainty. Accordingly, the existing 2-percent uncertainty can be allocated such that 1.4 percent is applied to provide sufficient margin to address the uprate to 2689 MWt, and 0.6 percent is retained in the analysis to still account for the power measurement uncertainty.

Second, some analyses employ a nominal initial condition power level. These analyses have either been evaluated or re-performed for the 1.4-percent increased power level. The results demonstrate that the applicable analysis acceptance criteria continue to be met at the 1.4-percent conditions.

It should be noted that separate amendments, LAR-286 for Unit 1 and LAR-158 for Unit 2, have been submitted that implement the use of Revised Thermal Design Procedure methodology. This methodology generates departure from nucleate boiling margin, which will support the 1.4-percent uprating using the LEFM system. In addition, for these types of analyses, it is shown that they still employ other conservative assumptions not affected by the 1.4-percent uprated power. Taken together, the use of the calculated 95/95 power measurement uncertainty and retention of conservative assumptions indicate that the margin of safety for these analyses would not be reduced.

Third, some of the analyses already employ an initial condition power level in excess of the proposed 2,689 MWt. These analyses were previously performed at a higher power level as part of prior plant programs. For these analyses, some of this available margin has been used to offset the 1.4-percent uprate. Consequently, the analyses have been evaluated to confirm that sufficient analysis margin exists to envelope the 1.4-percent uprate.

Fourth, some of the analyses are performed at zero-percent initial condition power conditions or do not actually model the core power level. Consequently, these analyses have not been re-performed since they are unaffected by the core power level.

3.2 LEADING EDGE FLOW METER

The power uprate is based on the use of the Caldon LEFM equipment for determination of main feedwater flow and the associated determination of reactor power through the performance of a daily calorimetric. The uprate is based on the Caldon LEFM Check System on Unit 1 and the Caldon LEFM CheckPlus System on Unit 2.

The Beaver Valley LEFM systems will be extensively tested and calibrated at Alden Research Laboratories, in site-specific piping configurations prior to their installation at BVPS. The accuracy with which the device has been calibrated is factored into the uncertainty analyses for the BVPS units.

Unit 1 LEFM

The LEFM ultrasonic flow meter consists of an electronic cabinet in the Process Controls Area and a measurement section (spool piece) located in the 26-inch main feedwater header line. The measurement section holds eight ultrasonic transducer assemblies that are secured in their own transducer housing, which forms the pressure boundary. Each transducer may be removed at full-power conditions without disturbing the pressure boundary. The LEFM uses acoustic energy pulses to determine the final feedwater mass flow rate. Transducers that transmit and receive the pulses are mounted in the LEFM spool piece at an angle of 45 degrees to the flow. The sound will travel faster when the pulse traverses the pipe with the flow and slower when the pulse traverses the pipe against the flow. The LEFM uses these transit times and time differences between pulses to determine the fluid velocity and temperature. The system uses a single digital system controlled by software to employ the ultrasonic transit time method to measure four-line integral velocities at precise locations with respect to the pipe center line. The system numerically integrates the four velocities measured according to the method described in Caldon's Topical Report ER-80P. Although its use for calorimetric input is not nuclear safety related, the system's software has been developed and will be maintained under a verification and validation (V&V) program. The V&V program has been applied to all system software and hardware, and includes a detailed code review. The mass flow rate is displayed on the local display panel and transmitted to the plant process computer for use in the calorimetric measurement. The feedwater mass flow rate is used to determine the reactor thermal output based on an energy balance of the secondary system.

The LEFM is an improved system for use in determining and monitoring feedwater flow in nuclear power plants. The LEFM provides on-line verification of the accuracy of the feedwater flow and temperature measurements upon which NSSS thermal power determinations are based. In addition, the LEFM provides a significant improvement in accuracy and an increase in reliability of flow and temperature measurements.

The improved accuracy of measurements of feedwater mass flow and temperature results in a total uncertainty of less than ± 0.6-percent of reactor thermal power. This is substantially more accurate than the typical ± 2-percent rated thermal power (RTP) assumed in the accident analyses, or that uncertainty currently obtainable with precision, venturi-based instrumentation.

The LEFM system measures the transit times of pulses of ultrasonic energy traveling along chordal acoustic paths through the flowing fluid. This technology provides significantly higher accuracy and reliability than flow instruments, which use differential pressure measurements; and temperature instruments, which use conventional thermocouple or resistance thermometers.

The LEFM indications of feedwater mass flow will be directly substituted for the venturi-based flow indication and the resistance temperature detector (RTD) temperature indications currently used in the plant calorimetric measurement calculation performed with the plant computer. The plant computer will then calculate enthalpy and thermal power as it does now. The venturi-based feedwater flow measurement will continue to be used for feedwater control and other functions that it currently fulfills.

Unit 2 LEFM

The LEFM CheckPlus System is identical to the LEFM Check System on Unit 1 except for the addition of four acoustic measurement paths placed perpendicular to the first four. The addition of these paths provides improved accuracy to better than ± 0.32-percent RTP and also provides system redundancy. The enhanced accuracy of the CheckPlus system is not being credited on Unit 2 to remain consistent with Unit 1. The LEFM is substantially more accurate than the typical ± 2-percent RTP assumed in the accident analyses, or that uncertainty currently obtainable with precision, venturi-based instrumentation. The Unit 2 LEFM electronic cabinet is located in the Process Controls Area.

The LEFM indications of feedwater mass flow will be directly substituted for the venturi-based flow indication and the RTD temperature indications currently used in the plant calorimetric measurement calculation performed with the plant computer, as on Unit 1. The plant computer will then calculate enthalpy and thermal power as it does now. The venturi-based feedwater flow measurement will continue to be used for feedwater control and other functions that it currently fulfills.

3.3 NUCLEAR STEAM SUPPLY SYSTEM DESIGN PARAMETERS

3.3.1 Introduction

The NSSS design parameters are the fundamental parameters used as input in the NSSS analyses. They provide the reactor coolant system (RCS) and secondary system conditions (temperatures, pressures, and flow) that are used as the basis for the NSSS analyses and evaluations. As part of the 1.4-percent increase in licensed core power from 2,652 MWt to 2,689 MWt, it was necessary to revise these parameters. The new parameters are identified in Table 3-1. These parameters have been incorporated, as required, into the applicable NSSS systems and components evaluations, as well as safety analyses, performed in support of the uprate.

3.3.2 Input Parameters and Assumptions

The NSSS design parameters are determined based on conservative inputs, such as a conservatively low thermal design flow (TDF) and bounding steam generator tube plugging (SGTP) levels, which yield primary- and secondary-side conditions that bound the way the plant operates. The TDF is conservatively low relative to the measured RCS flow.

The modified input assumptions include the NSSS power level of 2,697 MWt (2,689 MWt core power) and increased feedwater temperature, which corresponds to increased power. These were the only input assumptions that changed in the calculation of the NSSS design parameters. Section 3.3.3 shows the effects of these modified input assumptions on the NSSS design parameters.

3.3.3 Results of Parameter Cases

Table 3-1 summarizes the NSSS parameter cases that were developed and used as the basis for the uprating project. The Analyses of Record (30-percent tube plugging) design parameters are also shown for comparison purposes. A description of the two uprated cases follows.

Case 1 represents the conditions with the current reactor power, reactor vessel average temperature of 576.2°F, and a 30-percent SGTP level. It yields the lowest possible initial secondary-side steam generator steam temperature, steam pressure, and steam flow for the analyses.

Case 2 represents the uprated conditions with the current reactor vessel average temperature of 576.2°F and a 0-percent SGTP level. It yields the highest possible initial primary-side temperatures for the analyses as well as the maximum secondary-side steam generator steam temperature, steam pressure, and steam flow.

Case 3 represents the uprated conditions with the current reactor vessel average temperature of 576.2°F and a 30-percent SGTP level. It yields the lowest possible initial secondary-side steam generator steam temperature, steam pressure, and steam flow for the analysis.

The 1.4-percent uprate results in changes to some of the NSSS design parameters, compared to the parameters that form the current licensing basis. The changes include the following RCS temperatures:

- T_{hot} increased by 0.4°F
- T_{cold} decreased by 0.4°F

These small changes occur since the vessel average temperature (T_{avg}) is maintained at the current design value (576.2°F) while increasing the core power by 37 MWt to 2,689 MWt. The temperature changes reflect the additional temperature difference across the uprated core taking into account core bypass flow.

In addition, the uprate results in the following changes to the secondary-side parameters at 30-percent SGTP:

- T_{steam} decreased by 1.0°F
- P_{steam} decreased by 5 psi
- $\dot{M}_{steam}$ increased by 1.6 percent

These small changes occur based on a calculation of the steam generator and secondary-side performance resulting from the increased core power. As a result of greater power coming from the steam generator, a higher steam flow is required along with a reduced enthalpy difference between the steam exiting the steam generator and the feedwater entering the steam generator. This latter effect results in a lower steam temperature and pressure.

3.3.4 Conclusions

The various NSSS analyses and evaluations described in this document used the design parameters appropriate for the given analytical area.

Relative to the 30-percent plugging parameters at 100-percent rated power, the following limitations apply:

1) The steam generator tube plugging is limited to the current Analyses of Record or administrative limit in place for Unit 2.

2) The steam generator outlet pressure must be maintained at or above 760 psia in order to remain in compliance with the component design transients assumptions associated with the component code stress and fatigue analyses.

3) Continued compliance with the Technical Specification thermal design (or minimum measured) flow rate must be satisfied. It is anticipated that the thermal design (or minimum measured) flow will be challenged for an average plugging level of less than 30 percent for Units 1 and 2.

Table 3-1
NSSS Design Parameters for Beaver Valley Units 1 & 2 – 1.4% Uprating

OWNER UTILITY: FirstEnergy Nuclear Operating Company (FENOC)
PLANT NAME: Beaver Valley
UNIT NUMBER: 1 & 2

BASIC COMPONENTS			
Reactor Vessel, ID, in.	157	Isolation Valves	Yes
Core		Number of Loops	3
Number of Assemblies	157	Steam Generator	
Rod Array	17x17V5H	Model	51 (U1)/51M(U2)
Rod OD, in.	0.374	Shell Design Pressure, psia	1100
Number of Grids	6Z, 2I[1]	Reactor Coolant Pump	
Active Fuel Length, in.	144	Model/Weir	93A/Yes
Number of Control Rods, FL	48	Pump Motor, hp	6000
Internals Type	DLW/DMW	Frequency, Hz	60

	Current Parameters	1.4% Uprating	
THERMAL DESIGN PARAMETERS	30% SGTP	0% SGTP	30% SGTP
NSSS Power %	100	101.4	101.4
MWt	2660	2697	2697
10^6 Btu/hr	9076	9203	9203
Reactor Power MWt	2652	2689	2689
10^6 Btu/hr	9049	9175	9175
Thermal Design Flow, loop gpm	87,200	87,200	87,200
Reactor 10^6 lb/hr	99.4	99.5	99.5
Reactor Coolant Pressure, psia	2250	2250	2250
Core Bypass, %	6.5	6.5	6.5
Reactor Coolant Temperature, °F			
Core Outlet	614.6	615.1	615.1
Vessel Outlet	610.4	610.8	610.8
Core Average	580.2	580.3	580.3
Vessel Average	576.2	576.2	576.2
Vessel/Core Inlet	542.0	541.6	541.6
Steam Generator Outlet	541.8	541.3	541.3
Steam Generator			
Steam Temperature, °F	506.5[2]	519.0	505.5[2]
Steam Pressure, psia	721[2]	806	716[2]
Steam Flow, 10^6 lb/hr total	11.59	11.81	11.78
Feed Temperature, °F	437.5	439.3	439.3
Moisture, % max.	0.25	0.25	0.25
Tube Plugging Level (%)	30	0	30
Zero-Load Temperature, °F	547	547	547

HYDRAULIC DESIGN PARAMETERS	
Pump Design Point, Flow (gpm)/head (ft)	88,500/280
Mechanical Design Flow, gpm	101,400
Minimum Measured Flow, gpm/total	266,800

FOOTNOTES:
(1) Plus protective bottom grid.
(2) Steam conditions are limited to minimums of 760 psia and 512.3°F due to component design transient considerations.

3.4 DESIGN TRANSIENTS

3.4.1 Nuclear Steam Supply System Design Transients

The revised NSSS performance design conditions and the NSSS design transients applicable to the uprated conditions serve as primary inputs to the evaluation and analysis of the NSSS systems and components (reactor vessel, pressurizer, RCS hot and cold leg piping, reactor coolant pumps, and steam generators). Current primary- and secondary-side design transients were reviewed to determine their continued applicability for the revised design conditions.

The methodology used to determine continued acceptability of the NSSS design transients as they are included in the design specifications for the above noted NSSS components was to compare the parameter change during the transient as it is presently defined against what would be the expected parameter change for the revised operating conditions resulting from the uprating. This was primarily done by looking at what would be the parameter change from the beginning to the end of the transient. This could be done since the various transients are developed for a plant condition proceeding from one steady-state point to another (for example, a 10-percent step change proceeds from 100-percent to 90-percent power condition in the transient analysis). By comparing the parameter change (e.g., T_{hot} or steam generator steam temperature) between the design transient initial and final condition as would be expected for the uprated condition versus the parameter change reported in the component engineering specification, a determination can be made on the continued acceptability of the design transient. If the existing design transient parameter change is bounded by that expected for the uprating, the existing design transient is judged to be bounding and remains valid.

- Note that this effort is primarily done by reviewing steady–state initial and final conditions in the design transients. No re-analyses are performed because the existing design transient analyses have sufficient conservatism built into them to accommodate uprating. For example: Transients usually are analyzed using minimum reactivity feedbacks, resulting in larger parameter swings than actually occur.

- Initial conditions for design transient analysis are primarily based upon 106.5 percent of nominal power level rating to allow for conservatism due to the generic basis used in the various transient analyses.

- Conservative values are used for steam generator tube fouling and heat transfer coefficient.

- Conservative definition of the transient. For example, 10-percent step load decrease might actually be analyzed as a 12-percent step change, or a loss of load takes no credit for any control systems, reactivity feedbacks, and no reactor trips except for the high pressurizer level trip.

Because of these conservatisms, the existing design transient parameter variations during the transient portion are judged to have sufficient conservatism to negate the need to perform a transient re-analysis.

A review of the design conditions indicate that the full-power temperature values for T_{hot} and T_{cold} vary by less than or equal to 0.5°F from the current design values. The vessel average temperature (T_{avg}) is unchanged. Given the conservative assumptions used to develop the current design transients, a 0.5°F change in primary-side full-power temperatures is judged to be negligible in regards to their effect on the design transients. Therefore, the existing design transients continue to be valid and applicable, without modification, at the revised design conditions.

3.4.2 Auxiliary Equipment Design Transients

The review of the NSSS auxiliary equipment design transients was based on a comparison between the revised operating conditions shown previously in Section 3.3 and the parameters that make up the current auxiliary equipment design transients. A review of the current auxiliary equipment transients determined that the only transients potentially impacted by the power uprate are those temperature transients impacted by full-load NSSS operating temperatures, namely T_{hot} and T_{cold}. These transients are currently based on an assumed full-load NSSS worst-case T_{hot} of 630°F and worst-case T_{cold} of 560°F. These NSSS temperatures were originally selected to ensure that the resulting design transients would be conservative for a wide range of NSSS operating temperatures.

A comparison of the limiting 1.4-percent uprate NSSS design temperature values for T_{hot} and T_{cold} (610.8°F and 541.6°F, respectively) with the existing transient temperature values indicates that they are still well within the design. Therefore, the actual temperature transients (that is, the change in temperature from T_{hot} or T_{cold} dictated by the power uprate parameters to a lower auxiliary system-related temperature or vice versa) are less severe than the current design temperature transients. The 1.4-percent uprate, therefore, does not require any changes to these transients.

3.5 NUCLEAR STEAM SUPPLY SYSTEMS

This section presents the results of the evaluations and analyses performed in the NSSS area to support the revised design conditions provided previously in Section 3.3. The systems addressed in this section include fluid systems and control systems. The results and conclusions of each evaluation and analysis are presented within each subsection.

3.5.1 Reactor Coolant System

Various assessments were performed to demonstrate that the RCS design basis functions could still be met at the revised design conditions. The potential impact of the uprated conditions on the previous RCS functions are described below.

a. The core power increase will affect the total amount of heat transferred to the main steam system (MSS). Verification that the major components can support this normal heat removal function is addressed in Section 3.6.

b. During the second phase of plant cooldown, the residual heat removal system (RHRS) will be required to remove larger amounts of decay heat from the RCS. Section 3.5.4 of this report addresses the RHRS cooldown capability at uprated conditions.

c. The increased thermal power can change the transient response to the RCS to normal and postulated design basis events. The acceptability of the RCS with respect to control and protection functions is addressed in Section 3.5.6.

d. The RCS TDF does not change as the result of the uprate. The reduction in cold leg temperature due to the uprate can reduce pressurizer spray flow. The pressurizer spray capability was evaluated at the 0.4°F lower cold leg temperature, and the 600 gpm design basis pressurizer spray continues to be achievable for the 1.4-percent uprate.

e. Reactor coolant system design temperature and pressure of 650°F and 2485 psig continue to remain applicable for the uprate conditions (Table 3-1).

f. The pressurizer design temperature and pressure of 680°F and 2485 psig continue to remain applicable for the uprate conditions (Table 3-1).

g. The pressurizer relief requirements have not changed due to the uprate (Section 3.5.6). Therefore, the following parameters are not affected:

    1. Pressurizer relief tank sizing and setpoints
    2. Pressurizer relief valve sizing and discharge piping pressure drop
    3. Pressurizer relief valve inlet pressure drop
    4. Pressurizer surge line pressure drop

3.5.2 Safety Injection System

Following a loss-of-coolant accident, the safety injection system (SIS) operates to remove the stored and fission product decay heat from the reactor core such that fuel rod damage, to the extent that it would impair effective cooling of the core, is prevented.

The "active" part of the SIS consists of high head safety injection pumps, the refueling water storage tank (RWST), low head safety injection pumps, and recirculation spray pumps, with the associated valves, instrumentation, and piping.

The passive portion of the SIS is the accumulator vessels that are connected to each of the RCS cold leg pipes. Each accumulator contains borated water under nitrogen pressure, and automatically injects into the RCS when the RCS pressure drops below the operating pressure of the accumulators. The active portion of the SIS (injection pumps) injects borated water from the RWST into the reactor following a break in either the RCS or steam system piping to cool the core and prevent an uncontrolled return to criticality. The arrangement of the safety injection (SI) pumps provides safety injection flow at any RCS pressure up to the set pressure of the pressurizer safety valves.

The SIS is described in two phases; the injection phase and the recirculation phase. The injection phase provides emergency core cooling and additional negative reactivity immediately following a spectrum of accidents including LOCA by prompt delivery of borated water to the reactor vessel. The recirculation phases provides long-term post-accident cooling by recirculating water from the containment sump.

Uprating of the power level does not directly impact the SIS operation or equipment performance, because they are dictated by system resistance. The current flow performance of record was reviewed and confirmed that the current flows provide acceptable results.

3.5.3 Chemical and Volume Control System

The chemical and volume control system (CVCS) provides for boric acid addition, chemical additions for corrosion control, reactor coolant cleanup and degasification, reactor coolant makeup, reprocessing of water letdown from the RCS, and RCP seal water injection. During plant operation, reactor coolant flows through the shell side of the regenerative heat exchanger and then through letdown orifices. The regenerative heat exchanger reduces the temperature of the reactor coolant and the letdown orifices reduce the pressure. The cooled, low-pressure water leaves the reactor containment and enters the Auxiliary Building. A second temperature reduction occurs in the tube side of the letdown heat exchanger followed by a second pressure reduction due to the low-pressure letdown valve. After passing through one of the mixed bed demineralizers, where ionic impurities are removed, coolant flows through the reactor coolant filter and enters the volume control tank (VCT).

In the assessment of CVCS operation at revised RCS operating temperatures, the maximum expected RCS T_{cold} must be less than or equal to the applicable CVCS design temperature and less than or equal to the heat exchanger design inlet operating temperature. The former criterion supports the functional operability of the system and its components. The latter criterion confirms that the heat exchanger design operating conditions remain bounding.

As for the CVCS thermal performance, the maximum T_{cold} of 541.3°F is still lower than the regenerative heat exchangers design inlet operating temperature of 543.5°F for Unit 1 and 542.5°F for Unit 2. Also, it is much lower than the shell-side structural design temperature of 650°F for the regenerative heat exchanger. The excess letdown heat exchanger inlet temperature (541.3°F) is less than the design inlet operating temperature (543.5°F for Unit 1, 547°F for Unit 2), which results in a lower excess letdown outlet temperature. The excess letdown path is primarily used for RCP seal injection when normal letdown flow is not available. Generally, the RCP seals are sensitive to higher temperatures. Therefore, a lower seal water heat exchanger outlet temperature, leading to a lower seal injection fluid temperature, will not present a problem.

Resizing of CVCS equipment is not required to support the uprating. Evaluation of required ECCS water volumes and boric acid concentrations will be performed as part of the normal

Reload Safety Evaluation process. The slight increase of N-16 activity at uprate conditions has a negligible effect on letdown/excess letdown line delay time requirements.

3.5.4 Residual Heat Removal System

The RHRS is designed to remove sensible and decay heat from the core and to reduce the temperature of the RCS during the second phase of plant cooldown. As a secondary function, the RHRS is used to transfer refueling water between the RWST and the refueling cavity at the beginning and end of refueling operations.

The RHRS consists of two heat exchangers, two RHR pumps, and associated piping, valves, and instrumentation. During system operation, coolant flows from one hot leg of the RCS to the RHR pumps, through the tube side of the residual heat exchangers, and back to two of the three RCS cold legs. The RHR heat exchangers are of the shell and U-tube type. Reactor coolant circulates through the tubes, while component cooling water circulates through the shell.

In all of the cases listed below, the conservatism of no credit for heat removal via steam generator steaming is applied.

Unit 1 Evaluation

A cooldown of the reactor coolant system using the full residual heat removal, component cooling water, and river water systems' capability is a normal plant evolution. The licensing basis for Unit 1 safe shutdown is hot standby, and the normal cooldown rate is used for maintenance planning. The normal cooldown is based on the full utilization of two residual heat removal pumps, two component cooling water pumps, two river water pumps, two residual heat removal heat exchangers, and three component cooling water heat exchangers. The Origen computer code is used as the basis for the decay heat level. The cooldown time to a reactor coolant system temperature of 140°F is increased by 0.1 hours compared to the previous power level of 2,652 MWt.

Unit 2 Evaluation

The licensing basis for Unit 2 is stated as hot standby, but UFSAR assesses the cold shutdown compliance with U.S. NRC Reactor Systems Branch Technical Position 5-1 (The Updated Final Safety Analysis Report (UFSAR) Section 5.4.7 refers to a Regulatory Guide 1.139 basis cooldown, which was in draft form during Unit 2 licensing). The following cooldown cases were evaluated for their effect by the 1.4-percent uprate:

- Normal Cooldown

  A cooldown of the reactor coolant system using the full residual heat removal, component cooling water, and service water systems' capability is a normal plant evolution. The normal cooldown rate is used for maintenance planning. The normal cooldown is based on the full utilization of two residual heat removal pumps, two component cooling water pumps, two service water pumps, two residual heat removal heat exchangers, and three component cooling water heat exchangers. The Origen computer code is used as the basis for the decay heat level. The cooldown time to a reactor coolant system temperature of 140°F increased by less than 0.1 hours compared to the previous power level of 2,652 MWt.

- Single Train Cooldown

  The cooldown time with a single train of cooling equipment in service will be extended compared to the normal cooldown. The single train cooldown is based on the utilization of one residual heat removal (RHR) pump, one component cooling water (CCW) pump, one service water pump, one RHR heat exchanger, and one CCW heat exchanger. The Origen computer code is used as the basis for the decay heat level. The cooldown time to a reactor coolant system temperature of 140°F increases by 0.5 hours compared to the previous power level of 2,652 MWt.

- Regulatory Guide 1.139

  The Regulatory Guide 1.139 cooldown case presented in the UFSAR is based on achieving cold shutdown using only safety–related equipment and assumes a loss of offsite power and a single failure. The RHRS is aligned into service within 36 hours after shutdown after the initial cooldown using the atmospheric steam dump valves and residual heat removal valve (refer to Section 3.7.1 for the evaluation of these valves). The acceptance criterion is that the RHRS must be capable of full decay heat removal after cut-in and eventually achieving cold shutdown (200°F). This case is based on one residual heat removal train, one component cooling water pump, and two component cooling water heat exchangers in service. For the Regulatory Guide 1.139 case, the effect of the 1.4-percent uprate is to extend the cooldown time to 200°F by 0.1 hours.

3.5.5 Spent Fuel Pool Cooling System

The Unit 1 (Unit 2) fuel pool cooling system removes the decay heat generated by the stored spent fuel assemblies. The system consists of two 100-percent capacity pumps and heat exchangers that are designed for a temperature and pressure of 200°F and 150 psig, respectively.

The Unit 1 heat exchangers are designed to remove the decay heat load, up to and including a full core offload. Thermal cases evaluated in conjunction with core offload to generate maximum pool temperatures included consideration for single and dual train cooling. The thermal-hydraulic analysis that was originally performed to generate fuel pool temperatures was conducted at a core thermal power of 2,660 MWt.

The calculations were re-evaluated to include a 1.4-percent increase in core thermal power. Results of this analysis show that under all conditions, the fuel pool temperature increase is less than 2 degrees. This does not result in pool boiling under single failure. The time to boil with a complete loss of cooling is minimally affected.

The Unit 2 Spent Fuel Cooling System was also evaluated to demonstrate that the cooling system remains adequate for the design thermal-hydraulic cases. The thermal-hydraulic analysis was originally performed considering a thermal power of 2,714 MWt. This bounds the uprate thermal power, and is therefore acceptable.

3.5.6 NSSS Control Systems

Condition I transients are evaluated to confirm that the plant can appropriately respond to these transients without generating a reactor trip or engineered safety feature actuation system (ESFAS) actuation. The transients of concern include:

- 10-percent step load increase
- 10-percent step load decrease
- 50-percent load rejection
- 5-percent per minute ramp load increase

The analysis methodology for these transients employs a 2-percent power calorimetric uncertainty to increase the power level to 102 percent. The improved thermal power measurement accuracy obviates the need for the full 2-percent power measurement margin assumed in the analysis.

Furthermore, the power measurement margin is only one of many conservative assumptions used in the analysis. Others include a minimum available steam dump capacity and more limiting beginning-of-life (BOL) fuel reactivity conditions (which provide the more severe reactivity response, and hence transient conditions). Together, the improved power measurement uncertainty and conservative assumptions provide substantial conservatism such that the transients noted above can be accommodated without resulting in a reactor trip or ESFAS actuation. Previous analyses were assumed at a full-power level of 102 percent. The 102-percent power level bounds the proposed 1.4-percent uprate conditions. Therefore, the current analyses remain valid and bound the 1.4-percent uprating conditions.

Likewise, the pressurizer power-operated relief valve (PORV) and spray valve capacity for response to key operational transients is determined to be unaffected by the power uprate due to the use of a 2-percent power uncertainty and other conservatisms. The sizing basis for the pressurizer spray valves (which addresses spray flow) is to prevent challenging the pressurizer PORVs for a 10-percent step load decrease transient. Previous analyses of a 10-percent load decrease from 102-percent power have shown that the installed spray valves are adequate. The initial power level of 102 percent bounds the proposed 1.4-percent uprating. Therefore, the current spray size and flow are adequate. There have been no changes to the current design transients that will impact surge line flow. Therefore, surge line flows have not changed and remain adequate.

The rod and steam dump control system stability for key operational transients was also examined. They are not a function of power level or full-load T_{avg}, but rather a function of the rod and steam dump control system setpoints and the reactor core kinetics. Since the 1.4-percent uprating does not include any change to the control systems setpoints or represent any significant change in the reactor core kinetics, the rod and steam dump control system stability are not affected by the 1.4-percent uprating. A control room annunciator for LEFM trouble is planned and no other changes for controls or displays are required as a direct result of the power uprate. In conclusion, the 1.4-percent uprating does not result in changes to plant operating conditions that would require any control system setpoint modifications.

3.5.7 Cold Overpressure Mitigation System

The cold overpressure mitigation system (COMS) is designed to protect the RCS from overpressure events when the RCS temperature is below 329°F for Unit 1 and below 350°F for Unit 2. Changes to full-power operating parameters, such as NSSS power, do not impact COMS. Thus, the existing COMS analysis is unaffected, and the 10CFR50, Appendix G curves and reference temperature values do not change (refer to Section 3.6.2). It should be noted that the applicable P-T curves have been designated from 15 EFPY to 14 EFPY on Unit 2 to accommodate uprate. The P-T curves have not changed and thus the existing setpoints are not affected.

3.6 NUCLEAR STEAM SUPPLY SYSTEM COMPONENTS

3.6.1 Reactor Vessel Structural Evaluation

This evaluation assesses the effects that the 1.4-percent uprating conditions have on the most limiting locations with regard to ranges of stress intensity and fatigue usage factors in each of the vessel regions as identified in the reactor vessel stress reports and addenda.

The NSSS design transients are demonstrated to be unaffected by the 1.4-percent power uprate (see Section 3.4). However, the vessel outlet temperature increases from 610.4°F to 610.8°F and the vessel inlet temperature decreases from the current 542.0°F to 541.6°F as a result of the 1.4-percent uprate program. Therefore, both the T_{hot} and T_{cold} variation during normal plant loading and plant unloading are increased. Plant loading and unloading are considered to be the more severe transients in the reactor vessel evaluations, but the impacts are small and well within the reactor vessel margins on both Units 1 and 2 as discussed below.

Unit 1 Evaluation

The vessel outlet temperature associated with the 1.4-percent power uprate is less than the normal operating vessel outlet temperature that was originally analyzed for the Unit 1 reactor vessel outlet nozzles. Therefore, the effects of the plant loading and unloading transients on the outlet nozzles are bounded by the original reactor vessel stress report.

The reactor vessel main closure flange region and control rod drive mechanism (CRDM) housings were previously evaluated for the effects of a higher vessel outlet temperature. Therefore, the effects of the 1.4-percent uprate vessel outlet temperature on these regions are also bounded by the current design basis.

The remaining reactor vessel regions, including the inlet nozzles, vessel wall transition, core support guides, bottom head-to-shell juncture, and instrumentation tubes were evaluated to assess the impact of the vessel inlet temperature variation associated with the 1.4-percent power uprate. It is concluded that the small vessel inlet temperature variation during plant loading and unloading has no effect on either the maximum stress intensity range or the maximum cumulative fatigue usage factor for these regions.

The Code version used in the evaluation for Beaver Valley Unit 1 is the 1968 Edition of Section III of ASME B&PV Code through the Winter 1968 Addenda. The Code is the same as the current Code of Record for the respective components.

Unit 2 Evaluation

An evaluation of the Unit 2 reactor vessel outlet nozzles, main closure flange, and CRDM housings concluded that the maximum ranges of primary-plus-secondary-stress intensity and maximum cumulative fatigue usage factors reported for these regions are not affected by the small increase in the vessel outlet temperature.

The vessel inlet temperature associated with the 1.4-percent power uprate provides a temperature variation of 5.4°F during plant loading and unloading. This magnitude of temperature change is less than the 7.0°F change in T_{cold} considered for plant loading and unloading in the original reactor vessel stress report. Therefore, the effects of the revised T_{cold} variation during plant loading and unloading are considered to be bounded by the original analysis.

The maximum ranges of stress intensity and maximum cumulative fatigue usage factors for the Unit 2 reactor vessel remain unchanged when the 1.4-perecent power uprate parameters are considered.

The Code version used in the evaluation for Beaver Valley Unit 2 is the 1971 Edition of Section III of ASME B&PV Code through the Summer 1972 Addenda. The Code is the same as the current Code of Record for the respective components.

Conclusion

The Unit 1 and Unit 2 reactor vessel evaluations for the 1.4-percent power uprate demonstrate that the maximum ranges of stress intensity remain within their applicable acceptance criteria, and the maximum cumulative fatigue usage factors remain below the acceptance criterion of 1.0.

In addition, the faulted condition stress analyses for the Beaver Valley Units 1 and 2 reactor vessels do not change as a result of the 1.4-percent power uprate because no changes in the faulted condition reactor vessel/reactor internals interface loads or other faulted conditions are identified as a result of the uprating.

3.6.2 Reactor Vessel Integrity – Neutron Irradiation

The reactor vessel integrity analysis was evaluated for the 1.4-percent uprate by examining the revised design conditions (provided previously in Section 3.3) and the increase in neutron fluences. The current analyses assume that the T_{cold} is maintained between 530°F and 590°F. The T_{cold} of 541.6°F for the 1.4-percent uprate is within this range. Therefore, the temperature assumption for the analyses is not affected.

3.6.2.1 Unit 1 Evaluation

Neutron Fluence

Neutron fluence projections on the vessel were evaluated for the uprated power level. The fluence projections serve as input to the reactor vessel integrity evaluations. Specifically, fluence values are used to evaluate the end-of-life (EOL) transition temperature shift for development of the surveillance capsule withdrawal schedules, determine EOL upper shelf energy (USE) values, adjust reference temperature values for determining the applicability of the heatup and cooldown curves, adjust Emergency Response Guideline (ERG) limits, and determine pressurized thermal shock (RT_{PTS}) values.

An evaluation of the neutron exposure of the reactor vessel materials was performed to bound the effects of the proposed 1.4-percent increase in RTP. This evaluation includes assessments not only at locations of maximum exposure at the inner diameter of the vessel, but also as a function of axial, azimuthal, and radial location throughout the vessel wall.

The fast neutron exposure levels are defined at depths within the vessel wall equal to 25 and 75 percent of the wall thickness for each of the materials constituting the beltline region. This is done to satisfy the requirements of the Code of Federal Regulations (CFR) 10CFR50, Appendix G, for the calculation of pressure/temperature limit curves for normal heatup and cooldown of the reactor coolant system. These locations are commonly referred to as the 1/4T and 3/4T positions in the vessel wall. The 1/4T exposure levels are also used in the determination of upper shelf fracture toughness as specified in 10CFR50, Appendix G. Maximum neutron exposure levels experienced by each of the beltline materials are required for determining the RT_{PTS} values. These RT_{PTS} values are compared with the applicable pressurized thermal shock screening criterion as defined in 10CFR50.61. The maximum exposure levels occur at the vessel inner radius.

The result of the fast neutron exposure evaluation for Beaver Valley Unit 1 accounts for the uprated power level. The result is based on the conservative assumption that the power uprate is initiated coincident with the last surveillance capsule withdrawal (capsule Y) from the unit. The resulting fast neutron ($E > 1.0$ MeV) exposure projections increase due to the power uprate. The new projections were used as input to the reactor vessel integrity evaluations.

Surveillance Capsule Withdrawal Schedule

A withdrawal schedule is developed to periodically remove surveillance capsules from the reactor vessel to effectively monitor the condition of the reactor vessel materials under actual operating conditions. A surveillance capsule withdrawal schedule has been developed for the Beaver Valley Unit 1 reactor vessel based on the projected neutron fluence values resulting from the 1.4-percent power uprate. The surveillance capsule withdrawal schedule is based on American Society for Testing and Materials (ASTM) E185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels." The withdrawal of a capsule is scheduled at the nearest vessel refueling outage to the calculated effective full-power years (EFPYs). To date, four capsules have been removed and analyzed from the Beaver Valley Unit 1 reactor vessel. The next capsule to be removed is Capsule X at 25.7 EFPY.

Heatup and Cooldown Pressure - Temperature Limit Curves

An evaluation was performed at the uprated condition for the 16 EFPY heatup and cooldown curves and was determined to be bounded for the remaining applicability. Capsule Y was withdrawn in 1R13 Spring of 2000. A revised evaluation must be submitted within one year of capsule withdrawal per 10CFR50 Appendix H.

Pressurized Thermal Shock (PTS)

The RT_{PTS} screening criteria values were set (using conservative fracture mechanics analysis techniques) for beltline axial welds, plates, and beltline circumferential weld seams for end-of-license operation based on the NRC screening criterion for pressurized thermal shock (10CFR50.61). The RT_{PTS} values for beltline region materials of the Beaver Valley Unit 1 reactor vessel for end of license (28 EFPY) were recalculated and bound the 1.4-percent uprate. These RT_{PTS} values increase due to the 1.4-percent uprating. However, other circumstances such as updated chemistry factor values and the analysis of Capsule Y also have an effect on the results. The Beaver Valley Unit 1 RT_{PTS} values remain below the NRC screening criteria values using projected fluence values through 28 EFPY (current license).

Emergency Response Guideline Limits

New RT_{PTS} values were determined for Beaver Valley Unit 1 based on the bounding fluence projections. A comparison of the current RT_{PTS} calculation (which is the RT_{NDT} value at the end-of-license (28 EFPY)) to the uprated RT_{PTS} values for Beaver Valley Unit 1 was made to determine if the applicable ERG category (Westinghouse Owners Group Emergency Response Guidelines, Rev. 1C, September 30, 1997) would change.

The most limiting RT_{PTS} value for Beaver Valley Unit 1 is 259°F at 28 EFPY. The Beaver Valley Unit 1 limiting material is the lower shell plate B6903-1. The ERGs were developed for three specific categories. The first two categories were developed for an axial flaw in a longitudinal weld or plate or forging. The third category was developed for a circumferential weld flaw with an ART greater than 250°F. Beaver Valley Unit 1 would be in Category II through approximately 21.7 EFPY. The ERG limit for operation beyond 21.7 EFPY will need to be based on a plant-specific evaluation.

Upper Shelf Energy

Since the bounding neutron fluence values for the 1.4-percent uprate have increased, the USE values were recalculated for Beaver Valley Unit 1. It is determined that the reactor vessel beltline materials in the reactor vessel are expected to have a USE greater than 50 ft-lb through the end of license (28 EFPY) as required by 10CFR50, Appendix G.

3.6.2.2 Unit 2 Evaluation

Neutron Fluence

Neutron fluence projections on the vessel were evaluated for the uprated power level. The fluence projections serve as input to the reactor vessel integrity evaluations. Specifically, fluence values are used to evaluate the EOL transition temperature shift for development of the surveillance capsule withdrawal schedules, determine EOL USE values, adjust reference temperature values for determining the applicability of the heatup and cooldown curves, adjust ERG limits, and determine pressurized thermal shock values.

An evaluation of the neutron exposure of the reactor vessel materials to determine the effects of the 1.4-percent increase in core power was performed. This evaluation includes assessments not only at locations of maximum exposure at the inner diameter of the vessel, but also as a function of axial, azimuthal, and radial location throughout the vessel wall.

The fast neutron exposure levels are defined at depths within the vessel wall equal to 25 and 75 percent of the wall thickness for each of the materials constituting the beltline region. This is done to satisfy the requirements of 10CFR50, Appendix G, for the calculation of pressure/temperature limit curves for normal heatup and cooldown of the reactor coolant system. These locations are commonly referred to as the 1/4T and 3/4T positions in the vessel wall. The 1/4T exposure levels are also used in the determination of upper shelf fracture toughness as specified in 10CFR50, Appendix G. Maximum neutron exposure levels experienced by each of the beltline materials are required for determining the RT_{PTS} values. These RT_{PTS} values are compared with the applicable pressurized thermal shock screening criterion as defined in 10CFR50.61. The maximum exposure levels occur at the vessel inner radius.

The result of the fast neutron exposure evaluation for Beaver Valley Unit 2 accounts for the uprated power level. The result is based on the conservative assumption that the power uprate is initiated coincident with the last surveillance capsule (capsule V) withdrawal from the unit. The resulting fast neutron ($E > 1.0$ MeV) exposure projections increase due to the power uprate. The new projections were used as input to the reactor vessel integrity evaluations.

Surveillance Capsule Withdrawal Schedule

A withdrawal schedule is developed to periodically remove surveillance capsules from the reactor vessel to effectively monitor the condition of the reactor vessel materials under actual operating conditions. The current withdrawal schedules were evaluated based on the revised fluence projections. It was determined that no change to the current withdrawal schedules is necessary for Beaver Valley Unit 2.

Heatup and Cooldown Pressure – Temperature Limit Curves

An evaluation of the current 15 EFPY heatup and cooldown curves for Beaver Valley Unit 2 was performed to determine if a change in EFPY was required due to the uprating fluence values. The heatup and cooldown curves are documented in WCAP-15139, "Beaver Valley Unit 2 Heatup and Cooldown Limit Curves During Normal Operation at 15 EFPY Using Code Case N-626." The heatup and cooldown curves documented in WCAP-15139 were generated using the most limiting adjusted reference temperature (ART) ART values and the NRC-approved methodology documented in WCAP-14040-NP-A, Revision 2, with two exceptions. These exceptions are:

- The fluence values are calculated fluence values, not the best-estimate fluence values.
- The K_{Ic} critical stress intensities are used in place of the K_{Ia} critical stress intensities based on the approved methodology in ASME Code Case N-640.

The results of the evaluation of the current heatup and cooldown curves conclude that the curves are applicable to 14 EFPY given the 1.4-percent uprate conditions. Thus, a revision to the Technical Specification P-T curve from 15 EFPY to 14 EFPY is required to accommodate uprated conditions.

Pressurized Thermal Shock

The RT_{PTS} screening criteria values are set (using conservative fracture mechanics analysis techniques) for beltline axial welds, plates, and beltline circumferential weld seams for end-of-license operation based on the NRC screening criterion for pressurized thermal shock (10CFR50.61). The RT_{PTS} values for beltline region materials of the Beaver Valley Unit 2 reactor vessel for end of license (32 EFPY) were recalculated for the 1.4-percent uprate. These RT_{PTS} values increase due to the 1.4-percent uprating. The Beaver Valley Unit 2 RT_{PTS} values remain below the NRC screening criteria values using projected fluence values through 32 EFPY.

Emergency Response Guideline Limits

New RT_{PTS} values were determined for Beaver Valley Unit 2 based on the revised fluence projections for the 1.4-percent uprate. A comparison of the current RT_{PTS} calculation (which is the RT_{PTS} value at the end-of-license (32 EFPY)) to the uprated RT_{PTS} values for Beaver Valley Unit 2 was made to determine if the applicable ERG category (Westinghouse Owners Group Emergency Response Guidelines, Rev. 1C, September 30, 1997) would change.

The most limiting RT_{PTS} value for Beaver Valley Unit 2 is 154°F at 32 EFPY. Since this value is well below the 200°F maximum for Category I ERG limits, the Beaver Valley Unit 2 ERG plant-specific limits for current EOL (32 EFPY) remain valid for the 1.4-percent uprate.

Upper Shelf Energy

Since the neutron fluence values for the 1.4-percent uprate have increased, the USE values were recalculated for Beaver Valley Unit 2. It is determined that the reactor vessel beltline materials in the reactor vessel are expected to have a USE greater than 50 ft-lb through the end of license (32 EFPY) as required by 10CFR50, Appendix G.

3.6.3 Reactor Internals

The reactor internals support the fuel and control rod assemblies, absorb control rod assembly dynamic loads, and transmit these and other loads to the reactor vessel. The internals also direct flow through the fuel assemblies, provide adequate cooling to various internals structures, and support in-core instrumentation. The changes in the RCS design parameters, identified previously in Section 3.3, produce changes in the boundary conditions experienced by the reactor internals components. This section describes the evaluation performed to demonstrate that the reactor internals can perform their intended design functions at the 1.4-percent uprated conditions.

3.6.3.1 Thermal-Hydraulic Systems Evaluations

A key area in evaluation of core performance is the determination of hydraulic behavior of coolant flow and its effect within the reactor internals system. The core bypass flows are required to ensure reactor performance and adequate vessel head cooling. The hydraulic lift forces are critical in the assessment of the structural integrity of the reactor internals. Baffle gap momentum flux/fuel stability is affected by pressure differences between the core and baffle former region. The results of the thermal-hydraulic evaluations are provided below.

Core Bypass Flow Calculation

Bypass flow is the total amount of reactor coolant flow bypassing the core region and is not considered effective in the core heat transfer process. The principal core bypass flows are the barrel-baffle region, vessel head cooling spray nozzles, vessel outlet nozzle gap, baffle plate cavity gap, and the thimble tubes. An analysis demonstrated that the core bypass flow with the revised design conditions remains less than the current design value, and is therefore acceptable.

Hydraulic Lift Forces

The reactor internals hold-down spring is essentially a large-diameter Belleville-type spring of rectangular cross-section. The purpose of this spring is to maintain a net clamping force between the reactor vessel head flange and upper internals flange, and the reactor vessel shell flange and the core barrel flange of the internals. An evaluation demonstrated that the hydraulic lift forces on the various reactor internals components were enveloped by the current Analysis of Record. It is concluded, therefore, that the spring would maintain a net clamping force and the reactor internals assembly would remain seated and stable for the 1.4-percent power uprate conditions.

Baffle Joint Momentum Flux and Fuel Rod Stability

Baffle jetting is a hydraulically induced instability or vibration of fuel rods caused by a high-velocity jet of water. This jet is created by high-pressure water being forced through gaps between the baffle plates, which surround the core.

To minimize the propensity for flow-induced vibration, the crossflow emanating from baffle joint gaps must be limited to a specific momentum flux, V^2h; that is, the product of the gap width, h, and the square of the baffle joint jet velocity, V^2. This momentum flux varies from point to point along the baffle plate due to changes in pressure differential across the plate and the local gap width variations. In addition, the modal response of the vibrating fuel rod must be considered. That is, a large value of local momentum flux impinging near a grid is much less effective in causing vibration than the same V^2h impinging near the mid-span of a fuel rod.

The results show that for all modal shapes, the momentum flux does not change as a result of the 1.4-percent power uprate conditions.

Rod Cluster Control Assembly Drop Time Analyses

Technical Specification 3.1.3.4 for both Beaver Valley Units 1 and 2 requires that the rod cluster control assembly (RCCA) drop time be less than or equal to 2.7 seconds. The revised design conditions, in particular the reduced T_{cold}, can increase the drop time due to the increased fluid density. An evaluation confirmed that the RCCA drop time is still within the current value of 2.7 seconds at the revised design conditions. Therefore, the rod drop time remains valid for the 1.4-percent uprated conditions.

3.6.3.2 Mechanical Evaluations

The 1.4-percent uprate conditions do not affect the current design bases for seismic and LOCA loads. Therefore, it was not necessary to re-evaluate the structural effects from seismic operating-basis earthquake (OBE) and safe shutdown earthquake (SSE) loads, and the LOCA hydraulic and dynamic loads.

With regards to flow-and pump-induced vibration, the current analysis uses a mechanical design flow, which did not change for the revised design conditions. The revised design conditions slightly alter the T_{cold} and T_{hot} fluid densities, which slightly change the forces induced by flow. However, these changes are enveloped by the current Analysis of Record. In addition, T_{avg} is not changing as a result of the 1.4-percent uprate. Therefore, the mechanical loads are not affected by the 1.4-percent uprated conditions.

3.6.3.3 Structural Evaluations

Evaluations are required to demonstrate that the structural integrity of the reactor internal components is not adversely affected by the 1.4-percent uprate conditions. The presence of heat generated in reactor internal components, along with the various fluid temperatures, results in thermal gradients within and between components. These thermal gradients result in thermal stresses and thermal growth, which must be accounted for in the design and analysis of various components. The core support structures affected by the revised design conditions are discussed in the following sections. The primary inputs to the evaluations are the NSSS design parameters given previously in Section 3.3 and the gamma heating rates. The gamma heating rates were modified, as required, to account for the 1.4-percent increase in core power.

Baffle-Barrel Region Evaluations

The baffle-barrel regions consist of a core barrel into which baffle plates are installed, supported by bolting interconnecting former plates to the baffle and core barrel. The baffle-to-former bolts restrain the motion of the baffle plates that surround the core. These bolts are subjected to primary loads consisting of deadweight, hydraulic pressure differentials, and seismic loads, as well as secondary loads consisting of preload, and thermal loads resulting from RCS temperatures and gamma heating rates. The baffle-to-former bolt thermal loads are induced by differences in the average metal temperature between the core barrel and baffle plate. In addition to providing structural restraint, the baffles also channel and direct coolant flow so that a coolable core geometry can be maintained.

The thermal stresses in the core barrel shell in the core active region are primarily due to temperature gradients through the thickness of the core barrel shell. These temperature gradients are caused by the fluid temperatures between the inside and outside surfaces and the contribution of gamma heating.

A structural assessment determined that the 1.4-percent uprate conditions have no impact on the current Analysis of Record for the baffle plate and core barrel. No changes occur in the gamma heating rates for the baffle plate and core barrel. In fact, the new gamma heating rates for the baffle barrel region are significantly reduced due to the fuel low-leakage loading pattern being used in Beaver Valley Units 1 and 2. Thus, the ability to provide structural restraint and direct coolant flow (i.e., maintain coolable core geometry) of the baffle-barrel region is maintained.

Lower Core Plate Structural Analysis

The lower core plate is a perforated circular plate that supports and positions the fuel assemblies. The plate contains numerous holes to allow fluid flow through the plate. The fluid flow is provided to each fuel assembly and the baffle barrel region. The plate is bolted at the periphery to a ring welded to the inside diameter of the core barrel. The center span of the plate is supported by the lower support columns, which are attached at the lower end to the lower support plate.

Temperature differences between components of the lower support assembly induce thermal stresses in the lower core plate. In addition, due to the lower core plate's proximity to the core, and the thermal expansion of fuel rods at power, the heat generation rates in the lower core plate due to gamma heating cause a significant temperature increase in this component. Thermal expansion of the lower core plate is restricted by the lower support columns, lower support plate, and core barrel. These restraining items are exposed to the inlet temperature and have heat generation rates much lower than those found in the lower core plate.

Structural evaluations were performed to demonstrate that the structural integrity of the lower core plates is not adversely affected by the revised design conditions. It is determined that the calculated fatigue usage factor remains less than 1.0 and the lower core plate is, therefore, structurally adequate at the revised design conditions for the 1.4-percent power uprate.

3.6.4 Control Rod Drive Mechanisms

The Model L-106A CRDMs and the capped latch housings (CLHs) are installed in the Beaver Valley Units 1 and 2 reactor head. These components are affected by the reactor coolant pressure, vessel outlet temperature, and hot leg NSSS design transients.

According to Section 3.4, the current NSSS design transients remain unchanged for the 1.4-percent uprate program. In addition, the reactor coolant pressure (2250 psia) for the 1.4-percent uprate conditions remains the same as originally specified for the CRDMs and CLHs.

According to Section 3.3, the vessel outlet temperature for the 1.4-percent uprate has increased slightly to 610.8°F. An evaluation was performed to assess the impact of the temperature change on the CRDMs and CLHs. It is determined that the temperatures remain bounded by the applicable structural design analyses for these components.

Based upon the evaluation, it is concluded that the CRDMs and CLHs continue to meet structural design requirements for the 1.4-percent uprate.

3.6.5 Pressurizer Surge Line Piping

Parameters associated with the 1.4-percent uprating were reviewed for their impact on the design basis analysis for the Auxiliary Class 1 pressurizer surge line piping including the effects of thermal stratification. The following were considered in the evaluation: the NSSS design parameters, NSSS design transients, and changes at the reactor coolant loop Auxiliary Class 1 branch nozzle connections due to deadweight, thermal, seismic, and LOCA loading conditions.

The evaluation of pressurizer surge line stratification compared the change in T_{hot}, as reported in Section 3.3, to the current conditions. The small increase in T_{hot} is a benefit to surge line stratification since it reduces the delta-T between the pressurizer and the hot leg.

As discussed in Section 3.4, the NSSS design transients are not affected by the uprating. Therefore, the design transients remain valid for the pressurizer surge line piping.

There is no impact on the deadweight analysis due to the 1.4-percent uprate because there is no change in the weight of the Auxiliary Class 1 pressurizer surge line piping systems. The seismic response spectra remains unchanged. Therefore, there is no impact on the seismic analysis. Section 3.10.3 indicates that the 1.4-percent uprate conditions do not require a change to the LOCA hydraulic forcing functions. Therefore, there is no impact on the Auxiliary Class 1 branch nozzle connections due to deadweight, thermal, seismic, or LOCA loading conditions.

Based on the evaluation performed for the 1.4-percent uprate, the existing pressurizer surge line piping analysis remains valid.

3.6.6 Reactor Coolant Pumps and Motors

3.6.6.1 Reactor Coolant Pump Structural Analysis

The Model 93A RCPs are installed at Beaver Valley Units 1 and 2 in the cold leg of the reactor coolant loops. The RCPs are affected by the reactor coolant pressure, steam generator outlet temperature, and primary-side cold leg NSSS design transients. The maximum steam generator outlet temperature shown previously in Table 3-1 is 541.3°F. This temperature is lower than the design basis temperatures and, therefore, represents a less severe condition. Since the applicable NSSS design transients and the reactor coolant pressure are determined to be unaffected by the 1.4-percent uprate, the existing stress analyses are bounding and remain applicable for RCP pressure boundary components.

3.6.6.2 Reactor Coolant Pump Motor Evaluation

A previous Beaver Valley engineering study determined that the RCP motors were acceptable for continuous operation with limiting hot loop and cold loop conditions. The RCP motors were determined to remain acceptable for operation at the 1.4-percent uprate parameters based on the following:

- No-load T_{avg} is unchanged by this uprating. Therefore, the RCP hot start is not affected by the uprate.
- Limiting RCP motor starting conditions occur during RCS cold loop conditions that are not impacted by the 1.4-percent uprate.
- The RCP motor thrust bearing loads were evaluated and found to be acceptable for the 1.4-percent uprate parameters.

3.6.7 Steam Generators

3.6.7.1 Steam Generator Structural Integrity

As noted in Section 3.4, the NSSS design transients are demonstrated to be unaffected by the 1.4-percent uprate. These design transients were used as input to generate the original or baseline calculations. Since the operating conditions with the 1.4-percent and 30-percent steam generator tube plugging have slightly increased, scale factors were developed based upon the increase in operating conditions. The scale factors were applied to the baseline analysis results to develop revised stresses and fatigue usage.

The results of the structural evaluations indicate that all applicable fatigue usage values are still less than the allowable limit of 1.0. Therefore, the evaluations demonstrate that the steam generators meet the requirements of the American Society of Mechanical Engineers (ASME) Code limits for stress and fatigue for the 1.4-percent uprate conditions.

3.6.7.2 Steam Generator Thermal-Hydraulic Performance

The following evaluations and analyses were performed to assess the magnitude and importance of changes in the secondary-side thermal-hydraulic performance characteristics for the Beaver Valley Units 1 and 2 steam generators at the 1.4-percent power uprate conditions.

Circulation Ratio/Bundle Liquid Flow

The circulation ratio is a measure of tube bundle liquid flow in relation to the steam flow and is primarily a function of steam flow. The bundle liquid flow minimizes the accumulation of contaminants on the tubesheet and in the bundle. The 1.4-percent increase in power causes the bundle liquid flow to change by less than 1 percent decrease and the circulation ratio to decrease by less than 2 percent. Therefore, the uprating and other operating condition changes have minimal effect on this function. No effect on sludge accumulation or local concentrations is expected.

Damping Factor

The hydrodynamic stability of a steam generator is characterized by the damping factor. A negative value of this parameter indicates a stable unit, meaning that small perturbations of steam pressure or circulation ratio will diminish rather than grow in amplitude. An evaluation confirmed that the damping factor will remain at a highly negative value at the uprated design conditions. The steam generators will continue to remain hydrodynamically stable.

Steam Generator Mass

The 1.4-percent uprating of the plant, combined with an increase in tube plugging, will result in a change in the steam properties in the generator. A change in the steam/water balance results in a change in mass of the generator, that will then affect the dynamic response of the generator and its internals. The change in steam generator mass will result in a change in tube dynamic response/vibration. The change in fluid mass within the generator will also affect the thermal performance of the generator.

The proposed operating geometry after uprating includes a tube plugging limit of 30 percent. In going from 0-percent tube plugging at the nominal power level to 30-percent plugging at the 1.4-percent uprated power level, the overall steam generator mass will be reduced by approximately 2 percent. Note that the 1.4-percent uprating alone, without a change in the tube plugging limit, would essentially have no effect on the overall steam generator mass. The impact of the minor reduction of mass on thermal performance and tube fatigue was considered. It was concluded that the reduction in mass would have no quantifiable impact on either thermal performance or tube fatigue.

Steam Generator Pressure Drop

The increase in total secondary-side pressure drop resulting from the uprating is approximately 1 psi. This increase is very small in relation to the total feedwater system pressure drop and will have a negligible effect on the feedwater system operation.

Moisture Carryover

The performance of the Beaver Valley Units 1 and 2 moisture separator packages is primarily a function of steam flow, steam pressure, and water level. An analysis was performed to determine the effect of the power uprate on the Beaver Valley moisture carryover. This was accomplished by projecting the separator performance from field performance data for Beaver Valley. From the extrapolation of the field performance data, the moisture carryover is estimated to remain below 0.25 percent.

3.6.7.3 U-Bend Fatigue Evaluation

Changes in thermal and hydraulic conditions occur when a plant is uprated, including changes in circulation ratio, bundle liquid flow, void fraction, and U-bend fluid flow velocity. All of these changes are considered in the evaluation of U-bend vibration and fatigue, to properly address NRC Bulletin 88-02.

No additional steam generator tubes will need to be plugged to preclude the potential for U-bend fatigue. In addition, a preliminary assessment indicates that the existing 40-percent through wall plugging criterion for steam generator tubes will remain adequate. FENOC will perform a calculation to substantiate the adequacy of the plugging criterion.

An evaluation was performed to determine the impact that the revised design conditions associated with the 1.4-percent uprating had on the steam generator tube fatigue in the U-bend region. Key operating conditions used as input to the U-bend fatigue evaluation include steam flow, circulation ratio, and steam pressure. The evaluation focused on the most susceptible steam generator tubes in the plant. Based on the planned operating conditions with a minimum steam pressure of 760 psi, there are two tubes in Unit 2 and one tube in Unit 1 that will require plugging after an additional cycle of operation, due to fatigue considerations. The applicable tube for Unit 1 is Row 10, Column 53, in S/G "C" and the tubes for Unit 2 are Row 8, Column 60, in S/G "A" and Row 8, Column 69, in S/G "C". These tubes will be removed from service no later than the refueling following implementation of the 1.4-percent uprating.

3.6.7.4 Steam Generator Hardware Changes and Additions Evaluation

Evaluations were performed to determine the impact of the revised design conditions for the power uprate (shown previously in Section 3.3) on the structural integrity of the steam generator hardware changes and additions. These hardware changes and additions are qualified for installation in the Beaver Valley Units 1 and 2 steam generators. The first of these additions is the mechanical plug which are in both units. The other addition is the laser-welded sleeve for Unit 1 only. The steam generator hardware structural evaluations for the 1.4-percent uprated conditions were performed to the applicable requirements of the ASME Boiler and Pressure Vessel (B&PV) Code, Section III.

Steam Generator Tube Mechanical Plug

The Westinghouse tube mechanical plugs (i.e., "short" Alloy 600 and Alloy 690 versions and the "long" Alloy 690 version) were evaluated for the effects of changes to the thermal transients due to the power uprate.

The Westinghouse tube mechanical plug is adequately anchored in the tube for all steady-state and transient conditions. There is adequate friction to prevent dislodging of the plug and there is adequate leakage resistance for the limiting steady-state and transient loadings. All of the stress/allowable ratios are less than unity. This indicates that all primary stress limits are satisfied for the plug shell wall between the top land and the plug end cap. The plug shell continues to meet the Class 1 fatigue exemption requirements per Article N-415.1 of the 1966 Edition of Section III of the ASME Code, equivalent to NB-3222.4 of the 1989 Edition of the Code. Since the fatigue exemption requirements are satisfied, the usage factor will remain within the Code limit of 1.0 and an explicit calculation of the usage factor is not required.

Similarly, rolled Alloy 600 and 690 steam generator tube plugs manufactured by Framatome Technologies have been evaluated for the effects of changes to the thermal transients due to the power uprate. This evaluation demonstrates the continued adequacy of the Framatome rolled plugs to perform their intended function while fulfilling applicable ASME Boiler and Pressure Vessel Code Section III requirements.

Steam Generator Laser-Welded Sleeves (LWS)

The LWSs were evaluated for the effects of changes to the thermal transients due to the power uprate. The most important set of parameters for this evaluation was the 30-percent SGTP case (shown in Section 3.3). The NSSS design transients that are applicable for the uprated conditions are unchanged from the zero-percent SGTP. The maximum range of primary-plus-secondary stresses are comparable to the corresponding stress-range limits. The cumulative fatigue usage factor remains less than unity. The structural limits for pressure, stress-range, and fatigue continue to meet Section III of the ASME Code for the 1.4-percent power uprate conditions.

3.6.7.5 Inspection Program and Tube Repair Criteria

The applicable PCWG design parameters for the proposed 1.4-percent uprating for Beaver Valley Units 1 and 2 specify a minimum full-power steam pressure of 760 psia. This parameter remains unchanged from the value specified for operation prior to the uprating. Similarly, the design transient parameters (pressures and temperatures) in the component design specifications for the present operating conditions remain bounding for the conditions that will exist after uprating. On these bases, existing analyses to address steam generator structural integrity, that are based on the temperatures and pressures in the component design specifications, and that incorporate a minimum full power steam pressure of 760 psia, are unaffected by the uprating.

The Technical Specification plugging limit of 40-percent through wall is applied to anti-vibration bar (AVB) wear at both units, and to cold leg thinning at Unit 1. AVB wear rates for both of the Beaver Valley units are small. No tubes were reported with AVB wear depths greater than 40 percent through wall for either of the last two inspections performed at both units. The 1.4-percent uprate should have little or no effect on either AVB wear growth or cold leg thinning. Based on current wear rates, large structural margins are provided. Growth rate and thinning changes will be reviewed as additional inspection data becomes available.

The Beaver Valley steam generator program follows the inspection recommendations of the Electric Power Research Institute (EPRI) Steam Generator Inspection Guidelines, as well as inspection requirements of Generic Letter (GL) 95-05. While the criteria described by GL 95-05 is not implemented at Unit 2, the inspection is performed as if the criteria is applied. The only variance from this program is that for Unit 2, all distorted signal indications at tube support plate intersections, regardless of amplitude, are inspected using a rotating coil probe. Any Unit 2 tube support plate indication confirmed by rotating coil regardless of bobbin signal amplitude, will be repaired, until such time that the voltage based repair criteria is implemented. The full length bobbin program for both units is 100 percent of all tubes. Similarly, the hot leg top of the tubesheet region rotating coil probe inspection is 100 percent of all hot leg tubes. Inspection plans for identification of new degradation mechanisms are addressed through inspection program expansion plans, since all active tubes are currently inspected over their entire length.

With respect to the proposed 1.4-percent power uprate, the inspection program will include consideration of the higher temperatures in crack growth rate analyses. In the event that condition monitoring and operational assessments of inspection results indicate the need, expansion of inspection plans and repairs will be made. Observation of change in degradation growth rates will be incorporated into the operational assessment, and evaluated for association with potential effects related to the uprating.

3.6.8 Pressurizer

An analysis was performed to assess the impact of the revised NSSS parameters at the uprated conditions for Beaver Valley Units 1 and 2 on the pressurizer components. The conditions that could affect the primary-plus-secondary stresses, and the primary-plus-secondary-plus-peak stresses, are the changes in the RCS hot leg temperature (T_{hot}), the RCS cold leg temperature (T_{cold}), and the pressurizer transients. A review of the revised temperature parameters, provided previously in Section 3.3, shows that the changes in T_{hot} and T_{cold} are very small and are enveloped by the current stress analysis. Since the design transients (see Section 3.4) are also unaffected by the uprated conditions, the revised parameters do not impact the pressurizer stress and fatigue analysis. It is concluded that the pressurizer components meet the stress and fatigue analysis requirement of Section III of the ASME Code 1965 Edition, Winter 1966 Addenda for Unit 1, and the 1971 Edition, Summer 1972 Addenda for Unit 2, for plant operation at the 1.4-percent uprated conditions.

3.6.9 NSSS Auxiliary Equipment

The NSSS auxiliary equipment includes the heat exchangers, pumps, valves, and tanks in the auxiliary systems. An evaluation determined that the existing design conditions used in the fatigue analysis for these components envelop those reported previously in Section 3.3. The NSSS design transient evaluation presented in Section 3.4.1 also concludes that the power uprate design transients, which are applicable to the NSSS auxiliary valves, are bounded by the current design basis transients. The NSSS valves are those that are completely within the boundary of the NSSS and those that isolate the NSSS from the interfacing auxiliary system. Furthermore, as noted in Section 3.4.2, the current auxiliary equipment design transients, which apply to the auxiliary heat exchangers, pumps, tanks, and the remaining valves, remain applicable for the 1.4-percent power uprate conditions. Therefore, the components will continue to meet their current design criteria since the fatigue usage values for each component will still be less than the allowable limit of 1.0.

3.6.10 Fuel Assembly

The Beaver Valley Units 1 and 2 17x17 Vantage 5 Hybrid (V5H) fuel design was evaluated to determine the impact of the 1.4-percent uprate on the fuel assembly structural integrity. Since the core plate motions for the seismic and LOCA evaluations are not affected by the uprated conditions, there is no impact on the fuel assembly seismic/LOCA structural evaluation. The 1.4-percent uprate does not increase operating and transient loads such that they will adversely affect the fuel assembly functional requirements. Therefore, the fuel assembly structural integrity is not affected and the seismic and LOCA evaluations of the 17x17 V5H fuel design for Beaver Valley Units 1 and 2 is still applicable for the 1.4-percent uprate.

3.6.11 Leak Before Break

The current leak-before-break (LBB) analyses for Beaver Valley Units 1 and 2 justified the elimination of large primary-loop pipe rupture and the pressurizer surge line pipe rupture as the structural design basis. In order to demonstrate acceptability of the elimination of RCS primary loop pipe breaks and the pressurizer surge line breaks, the following objectives must be achieved:

- Demonstrate that margin exists between the "critical" crack size and a postulated crack that yields a detectable leak rate

- Demonstrate that there is sufficient margin between the leakage through a postulated crack and the leak detection capability

- Demonstrate margin on applied load

- Demonstrate that fatigue crack growth is negligible

These objectives were met in the Beaver Valley Units 1 and 2 LBB analyses.

As indicated in Sections 3.6.5 and 3.8.11, there is no impact on the loads of the reactor coolant loop piping and the pressurizer surge lines due to the power uprate conditions. The effect on material properties due to the slight changes in temperature will have insignificant impact on the LBB margins. Therefore, the current LBB analyses remain applicable for the 1.4-percent power uprate conditions.

3.6.12 Loop Stop Isolation Valves

The loop stop isolation valves (LSIVs) are installed in both the cold leg and the hot leg of the RCS. The evaluations for the LSIVs were performed for the more severe hot leg conditions, including the hot leg transients. The LSIVs evaluations are based on the vessel outlet reactor coolant temperatures. Although the vessel outlet temperatures increased slightly to 610.8°F for the 1.4-percent uprate, the vessel outlet temperatures evaluated in the design basis analyses were higher. Therefore, it is concluded that the previous analyses performed on the Beaver Valley LSIVs remain applicable for the 1.4-percent uprate conditions.

3.7 NSSS/BOP FLUID SYSTEMS INTERFACE

The following BOP fluid systems were reviewed to assess compliance with NSSS/BOP interface guidelines at the revised design conditions shown previously in Section 3.3.

3.7.1 Main Steam System

The following subsections summarize the evaluation of the major steam system components relative to the revised design conditions for the 1.4-percent power uprate. The major components of the main steam system (MSS) include the steam generator main steam safety valves (MSSVs), the steam generator atmospheric steam dump valves (ASDVs), and residual heat release control valve (RHRCV). Other major MSS components are the main steam isolation valves (MSIVs) on Unit 2 and the main steam non-return valves/trip valves (MSNVs/TVs) on Unit 1.

Steam Generator Main Steam Safety Valves

The MSSVs must have sufficient capacity so that main steam pressure does not exceed 110 percent of the steam generator shell-side design pressure (the maximum pressure allowed by the ASME B&PV Code) for the worst-case loss-of-heat-sink event. Based on this requirement, a conservative criterion was applied that the valves should be sized to relieve

100 percent of the maximum calculated steam flow at an accumulation pressure not exceeding 110 percent of the design pressure.

Each Beaver Valley unit has 15 safety valves with a total capacity of 12.785 x 10^6 lb/hr for Unit 1 and 12.727 x 10^6 lb/hr for Unit 2. These capacities are at the highest safety valve setpoint plus accumulation pressure. For Unit 1, this provides about 108 percent of the maximum calculated steam flow of 11.81 x 10^6 lb/hr for the revised design conditions. For Unit 2, this provides approximately 108 percent of the maximum calculated steam flow of 11.81 x 10^6 lb/hr for the power uprate conditions. Therefore, based on the range of NSSS performance parameters for the uprating, the capacity of the installed MSSVs meets the sizing criterion.

The excessive cooldown event assumes a maximum flow limit of 890,000 lb/hr at 1100 psia for each MSSV (as well as each steam generator ASDV, RHRCV, and each condenser steam dump valve). Since the actual capacity of any single MSSV, ASDV, RHRCV, or condenser steam dump valve is less than the maximum flow limit per valve, the maximum capacity criterion is satisfied.

Steam Generator Atmospheric Steam Dump Valves and Residual Heat Release Control Valves

The primary function of the ASDVs is to provide a means for decay heat removal and plant cooldown by discharging steam to the atmosphere when either the condenser, the condenser circulating water pumps, or steam dump to the condenser is not available. Under such circumstances, the ASDVs, in conjunction with the auxiliary feedwater system (AFWS), permit the plant to be cooled down from the pressure setpoint of the lowest-set MSSVs to the point where the RHRS can be placed in service. During cooldown, the ASDVs are either automatically or manually controlled. In automatic, each ASDV proportional and integral (P&I) controller compares steam line pressure to the pressure setpoint, which is manually set by the plant operator.

In the event of a tube rupture event in conjunction with loss of offsite power, the ASDVs are used to cool down the RCS to a temperature that permits equalization of the primary and secondary pressures at a pressure below the lowest-set MSSV. RCS cooldown and depressurization are required to preclude steam generator overfill and to terminate activity release to the atmosphere.

In addition, each unit has an RHRCV that is mounted on a header and serves all three steam generators through connections on each main steam line upstream of the MSIVs on Unit 2 and upstream of the MSNV/TVs on Unit 1. The valve is manually positioned from the main control room and the primary function is to augment the normal cooldown function of the ASDVs.

The steam generator ASDVs on Unit 1 and the ASDVs on Unit 2 in conjunction with the RHRCV on Unit 2 are sized to have a capacity equal to about 10 percent of the steam flow used for plant design, at no-load steam pressure. This sizing is compatible with normal cooldown capability and minimizes the water supply required by the AFWS.

In addition, the Unit 2 RHRCV and two out of three of the Unit 2 ASDVs afford redundancy such that either the RHRCV or two out of three of the ASDVs will permit a plant cooldown to RHRS operating conditions in 36 hours after shutdown. This sizing is compatible with the plant design relative to safety-grade cold shutdown.

The Unit 1 ASDVs have a total design capacity of 1,186,000 lb/hr at 1020 psia. The Unit 2 ASDVs, in conjunction with the RHRCV, have a total capacity of 1,383,148 lb/hr at 1020 psia. For the revised design conditions, the Unit 1 ASDV capacity is approximately 10 percent of the required maximum steam flow.

The Unit 2 ASDV capacity, in conjunction with the capacity of the RHRCV, is about 11.7 percent of the required maximum steam flow. The evaluation also confirms that either two out of three Unit 2 ASDVs or the Unit 2 RHRCV have adequate capacity at the power uprate conditions to satisfy the cooldown requirements dictated by safety-grade cold shutdown per UFSAR Appendix 5A. Since the design capacity of the installed ASDVs and the Unit 2 RHRCV meets the sizing criterion, the values are adequately sized for the 1.4-percent uprated conditions.

Main Steam Non-Return Valves/Trip Valves, Main Steam Isolation Valves, Main Steam Isolation Bypass Valves, and Main Steam Trip Bypass Valves

The MSNV/TVs on Unit 1 and the MSIVs on Unit 2 are located outside the containment and downstream of the MSSVs. The valves function to prevent the uncontrolled blowdown of more than one steam generator and to minimize the RCS cooldown and containment pressure to within acceptable limits following a main steam line break. To accomplish this function, the MSNV/TVs and MSIVs must be capable of an overall closure time of 8 seconds for Unit 1 (7 seconds for Unit 2). These requirements are not impacted by power uprate.

The MSIV bypass valves (Unit 1) and main steam trip bypass valves (Unit 2) are used to warm up the main steam lines and equalize pressure across the MSIVs and trip valves prior to opening the MSIVs bypass valves and main steam trip valves. The MSIV bypass valves and main steam trip bypass valves perform their function at no-load and low-power conditions where power uprate has no significant impact on main steam conditions (e.g., steam flow and steam pressure). Consequently, power uprate has no impact on the interface requirements for the MSIV bypass valves and main steam valves.

3.7.2 Condenser Steam Dump System

The steam dump system creates an artificial steam load by dumping steam from ahead of the turbine valves to the main condenser. The sizing criterion recommends that the steam dump system (valves and pipe) be capable of discharging 40 percent of the rated steam flow at full-load steam pressure to permit the NSSS to withstand an external load reduction of up to 50 percent of plant-rated electrical load without a reactor trip. To prevent a trip, the transient requires all NSSS control systems to be in automatic, including the reactor control system, which accommodates 10 percent of the load reduction. Following a reactor trip from full power, a steam dump capacity of 40 percent of rated steam flow will also prevent the MSSVs from lifting.

Steam Dump System Major Components

Each operating unit at Beaver Valley is provided with 18 condenser steam dump valves. The Unit 1 valves provide a total steam dump capacity of 10.07 x 10^6 lb/hr at 790 psia inlet pressure and the Unit 2 valves provide a total capacity of 10.55 x 10^6 lb/hr at 790 psig inlet pressure. The respective Unit 1 and Unit 2 total capacities provide steam dump capabilities of about 73.9 percent and 70.2 percent of the uprated guaranteed steam flow (11.80 x 10^6 lb/hr), at a full-load steam generator pressure of 760 psia, versus the sizing criterion of 40 percent of rated steam flow.

Operation of the NSSS within the proposed range of operating parameters at lower steam generator pressures and increased steam flows will not result in a steam dump capacity below the sizing criteria. Therefore, the condenser steam dump capacity is adequate for the 1.4-percent power uprate.

3.7.3 Condensate and Feedwater System

The condensate and feedwater system (C&FS) must automatically maintain steam generator water levels during steady-state and transient operations. The range of NSSS performance parameters results in a required feedwater volumetric flow increase of up to 1.8 percent during full-power operation. The higher feedwater flow has an impact on system pressure drop, which may increase by as much as 3.6 percent. The system has been evaluated to accommodate the system pressure drop for uprate.

The major components of the C&FS are the main feedwater isolation valves (Unit 2), the main feedwater control valves, and the C&FS pumps.

Main Feedwater Isolation Valves/Main Feedwater Control Valves

The Unit 2 main feedwater isolation valves (MFIVs) are located outside containment and downstream of the main feedwater control valves (MFCVs). The valves function in conjunction with the primary isolation signals to the MFCVs and back up trip signals to the feedwater pumps to provide redundant isolation of feedwater flow to the steam generators following a steam line break or a malfunction in the steam generator level control system. Isolation of feedwater flow is required to prevent containment overpressurization and excessive RCS cooldowns. To accomplish this function, the MFCVs on Unit 1 and the Unit 2 MFCVs and MFIVs must be capable of an overall closure time of 10 seconds for Unit 1 (7 seconds for Unit 2). These requirements are not impacted by power uprate.

Condensate and Feedwater System Pumps

The C&FS available head, in conjunction with the MFCV characteristics, must provide sufficient margin for feed control to provide adequate flow to the steam generators during steady-state and transient operation. A continuous steady feed flow should be maintained at all loads. To assure stable feedwater control, with constant speed feedwater pumps, the pressure drop across the MFCVs at rated flow (100-percent power) should be approximately equal to 1.5 times the feedwater system dynamic losses from the feed pump discharge through the steam generators. In addition, adequate margin should be available in the MFCVs at full-load conditions to permit a C&FS delivery of 96 percent of rated flow with a 100-psi pressure increase above the full-load pressure with the MFCVs fully open.

For the range of revised NSSS performance parameters for the uprate, the steam generator pressure variations and feedwater flow variations impact MFCV lift at full load. For the power uprate, the MFCV lift at full load is not expected to exceed 77 percent for either unit (refer to Section 3.8).

For steady-state feedwater control, a full-load MFCV lift is acceptable up to approximately 85 percent. Therefore, the Unit 1 and 2 MFCVs are acceptable for steady-state control.

To provide effective control of flow during normal operation, the MFCVs are required to stroke open or closed in 20 seconds over the anticipated inlet pressure control range (approximately 0 - 1600 psig). These requirements are not impacted by power uprate.

3.7.4 Auxiliary Feedwater System

The AFWS supplies feedwater to the secondary side of the steam generators at times when the normal feedwater system is not available, thereby maintaining the steam generator heat sink. The system provides feedwater to the steam generators during normal unit startup, hot standby, and cooldown operations and also functions as an engineered safeguards system. In the latter function, the AFWS is required to prevent core damage and system overpressurization during transients and accidents, such as a loss of normal feedwater or a secondary-system pipe break. The minimum flow requirements for the AFWS are dictated by accident analyses, and since the uprating does impact safety analyses performed at the nominal 100-percent power rating, evaluations were performed to confirm that the AFWS performance is acceptable at the uprated conditions. These evaluations are described in Section 3.10.4 of this report and show acceptable results.

Primary Plant Demineralized Water Storage Tank Requirements

The AFWS pumps for each Beaver Valley Unit are normally aligned to take suction from the primary plant demineralized water storage tank (PPDWST). To fulfill the engineered safety features (ESF) design functions, sufficient feedwater must be available during transient or accident conditions to enable the plant to be placed in a safe shutdown condition.

The limiting transient with respect to PPDWST inventory requirements is the loss-of-offsite-power (LOOP) transient. The Beaver Valley Unit 1 and Unit 2 licensing basis dictates that in the event of a LOOP, sufficient PPDWST usable inventory must be available to bring the unit from full-power to hot standby conditions, and maintain the plant at hot standby for 9 hours. In light of these design bases requirements, the Beaver Valley Unit 1 and Unit 2 PPDWST is designed to accommodate a Technical Specification minimum usable inventory of 140,000 gallons and 127,500 gallons, respectively. The minimum usable inventory is based on reactor trip from 102 percent of engineered safeguards design rated (ESDR) power. Since the proposed power uprate is based on improved calorimetric error, no change in the required inventory or the plant Technical Specifications is required for operation at the uprated power level.

3.7.5 Steam Generator Blowdown System

The steam generator blowdown system is used in conjunction with the chemical addition and sampling systems to control the chemical composition of the steam generator shell water within the specified limits. The blowdown system also controls the buildup of solids in the steam generator water.

The blowdown flowrates required during plant operation are based on chemistry control and tube-sheet sweep requirements to control the buildup of solids. The rate of addition of dissolved solids to the secondary systems is a function of condenser leakage and the quality of secondary makeup water, and the rate of generation of particulates is a function of erosion-corrosion (E/C) within the secondary systems. Since neither condenser leakage nor the quality of secondary makeup water is expected to be impacted by power uprate, the rate of blowdown required to address dissolved solids should not be impacted by power uprate. Theoretically, the potential for E/C increases with any increase in secondary system flowrates that may result from the increased flows at uprate. However, the overall effect of the minor increases in secondary system velocities is not expected to alter the E/C rates appreciably. Therefore, the required blowdown to control secondary chemistry and particulates will not be significantly impacted by power uprate.

The present range of NSSS operating parameters permits a maximum decrease in steam pressure from no load to full load of 260 psi (i.e., from 1,020 psia to 760 psia). Since the inlet pressure to the steam generator blowdown system varies proportionally with operating steam pressure, the blowdown flow control valves must be designed to handle a corresponding range of inlet pressures. Based on the revised range of NSSS parameters for power uprate, the no-load steam pressure (1,020 psia) remains the same and the full-load minimum steam pressure (760 psia) is within the present operating range. Therefore, the range of operating parameters revised for power uprate will not impact blowdown flow control.

3.8 BALANCE-OF-PLANT SYSTEMS

3.8.1 Heat Balance

Two heat balances were developed for each unit: one for the current NSSS power level of 2,660 MWt and one for the proposed uprate NSSS power level of 2,697 MWt. The BOP systems were evaluated for the uprate conditions using the data from these heat balances.

3.8.2 Condensate System and Condenser

The primary function of the condensate system is to supply preheated condensate, via the feedwater heater trains, to the suction of the steam generator feedwater pumps. The condensate system pressure, temperature, and flow rate will change slightly at the uprate power level. However, these parameters will still remain below the system and component design conditions. The condensate pumps have sufficient margin to continue to satisfy feed pump flow rate and net positive suction head requirements at the uprated conditions.

Steam flow to each condenser will increase as a result of the power uprate. However, the uprate conditions are bounded by the condenser design.

3.8.3 Feedwater System

The feedwater system supplies heated feedwater to the steam generators under all load conditions maintaining level within the programmed band. Level is maintained by positioning the feedwater control valve in the feedwater line to each steam generator.

For the power uprate, the feedwater flow rate will increase slightly for each unit but, will remain below system design capabilities. Feedwater flow rate and velocity through the feedwater heaters will also increase accordingly as a result of the power uprate but remain within the design of the components.

3.8.4 Extraction Steam System

The extraction steam system transmits steam from the high- and low-pressure main turbines to the shellside of the feedwater heaters for feedwater heating. During normal operation, steam from the high-pressure turbine is used to heat feedwater flowing through the first and second point heaters, and steam from the low-pressure turbines is used to heat feedwater flowing through the third, fourth, fifth, and sixth point heaters.

Implementation of uprate will yield greater extraction steam pressures, temperatures, and, in most cases, flows as indicated on the uprate heat balances. However, the uprate extraction steam conditions are bounded by the extraction steam system design. The flow velocities at current and uprate conditions are within equipment design limits and will not appreciably increase flow-accelerated corrosion relative to existing levels. Additionally, the extraction steam system is capable of precluding turbine water induction and minimizing the effects of flashing extraction steam on turbine overspeed at uprate conditions.

3.8.5 Heater Drains System

The Units 1 and 2 heater drain system (SD/HDS) and associated equipment were evaluated to ensure the ability of the system to function under power uprate conditions. SD (HDS) design parameters were reviewed and compared against power uprate conditions to determine that acceptable design margin exists for operation at uprate conditions. Additionally, a walkdown of both units was performed to establish existing feedwater heater level control valve position (i.e., percent open) under baseline conditions.

Pressures and temperatures associated with the power uprate for both Units 1 and 2 will remain bounded by the existing designs of the SD (HDS) systems and its components. SD (HDS) components will remain capable of passing additional flow rate associated with the power uprate conditions and component velocities will not exceed accepted maximum values.

3.8.6 Circulating Water System

The Circulating Water System (CWS) is a closed-loop system that provides cooling water for the main condenser of the turbine generator unit. The total design circulating water flow rate to each cooling tower is approximately 507,400 gpm.

The CWS system flow will remain essentially unchanged following power uprate. The increased levels of rejected heat, from an increase in turbine exhaust flow, will increase the CWS outlet temperature by approximately 0.5°F. The heat load under power uprate conditions will result in a slight backpressure increase in the condenser. However, the increased backpressure will remain within acceptable limits. The increase in outlet temperature, due to the increased heat load, is bounded by the CWS system design and can be accommodated by the cooling tower. A slight increase in evaporation rates can also be expected, requiring an increase in makeup rates under maximum summer conditions (less than a 2-percent increase). This slight increase is within the capability of the makeup supply from the river/service water system. The condenser vacuum system and steam jet air ejectors will also continue to support reliable plant operation at uprate. No modifications to the CWS or its components are required for a power uprate.

3.8.7 Reactor/Primary Component Cooling Water Systems

The reactor (primary) component cooling water (CCW) systems provide an intermediate cooling loop for removing heat from reactor plant auxiliary systems and transferring it to the river water (Unit 1) or service water (Unit 2) systems.

During normal operation, two/one CCW pumps (Unit1/Unit2) and two CCW heat exchangers transfer the design heat load from the components served. The CCW system is designed to supply 100°F water to the components cooled under all modes of operation except during RHR cooldown.

The CCW systems will continue to remove the required heat loads under normal conditions without exceeding their design temperature limits at uprate. Since the heat load increase due to the uprate is small, no modifications or changes in flow rates and operating limits are required.

3.8.8 River/Service Water Systems

The Units 1 and 2 river/service water systems provide cooling water to various safety-related and non-safety-related equipment. The power uprate will slightly increase the heat rejection to the river water system. However, the river/service water systems design pressure and temperature will not be exceeded by the uprate. System hydraulic analyses have been reviewed and show that adequate margin will remain following uprate. Increased heat loads from the primary CCW heat exchangers will occur as a result of the increase in spent fuel pool cooling loads and RHR cooldown loads. The increase in heat loads will have an insignificant effect on the river/service water systems. The river/service water system heat removal requirements for the recirculation spray heat exchangers at uprate conditions are bounded by existing analysis.

Although the river/service water systems will experience slightly higher heat loads during normal operation following the uprate, the existing systems will continue to satisfy its normal and accident functions with no modifications being required to the systems.

3.8.9 Turbine/Secondary Component Cooling Water Systems

The turbine/secondary component cooling water systems provide an intermediate cooling loop for removing heat from the turbine plant auxiliary systems and transferring it to the river water (Unit 1) or service water (Unit 2) systems. The systems remove heat from designated non-safety-related turbine plant components. The turbine/secondary component cooling water systems were evaluated to determine the impact due to the uprate. The results of the evaluation showed that the power uprate will slightly increase the system heat load for these systems. However, these changes are bounded by the systems designs.

3.8.10 Containment Depressurization

The containment depressurization system is composed of two groups of subsystems: the quench spray subsystems and the recirculation spray subsystems. These systems are designed to provide the necessary cooling and depressurization of the containment following a LOCA. The LOCA analyses have been performed at a power level that bounds the core power uprate, reference section 3.11.1.2. Therefore, the containment depressurization system, including long-term containment sump water cooling via the recirculation spray heat exchangers, is not impacted by the uprate.

3.8.11 Piping, Pipe Supports, and Pipe Whip

The piping systems evaluated for power uprate effects included the reactor coolant (including primary loop piping, primary equipment nozzles, primary equipment supports, and auxiliary piping), main steam, feedwater, high-pressure heater drains, CCW, and fuel pool cooling piping systems. The evaluations performed have concluded that these piping systems remain acceptable and will continue to satisfy design basis requirements in accordance with applicable design basis criteria, when considering the temperature, pressure, and flow rate effects resulting from the power uprate conditions. Specifically, Beaver Valley Unit 1 piping and related support systems remain within allowable stress limits in accordance with American National Standards Institute (ANSI) B31.1, 1967 edition, including the 1971 Addenda. Beaver Valley Unit 2 piping and related support systems remain within allowable stress limits in accordance with ASME Section III, 1971 edition, including Addenda through the Winter 1972 for Class 1, 2, and 3 piping, and ANSI B31.1, 1967 edition, including Addenda through June 30, 1972 for Class 4 piping. The evaluations also concluded that no piping or pipe support modifications are required as a result of the increased power level.

The evaluation also evaluated the effects of the uprate on pipe break, jet, and whip. Due to the resulting small increase in pipe stresses, no new postulated pipe break locations were identified in high-energy piping. In addition, since the uprate only results in a small increase in pressure, no significant increase in jet impingement loading or pipe whip forces will be experienced.

No specific assumptions were used for the piping system evaluations for power uprate. The methodology used in the piping system evaluations involved reviewing existing piping system stress levels to ensure that adequate design margin was available to accommodate the effects resulting from power uprate.

3.8.12 Turbine Generator

The capability of the Turbine Generators to perform at the proposed uprated power conditions was evaluated by the manufacturer, Siemens Westinghouse in a feasibility study published November 17, 2000. The review included the throttles, high-pressure and low-pressure turbines, the generators and exciters, as well as associated auxiliary equipment including moisture separator reheaters (MSRs) and relief valves. All turbine generator components were determined to have sufficient margin to enable operation at the uprated power conditions without requiring equipment modifications.

The existing turbine missile analysis was reviewed for the uprated power level. The effects of the uprated steam conditions on rotor disc temperatures were determined, and found to be equal to, or slightly less than the original disc design operating temperatures. Therefore, it is concluded that the proposed uprating is bounded by the existing turbine missile analysis.

The HP turbine impulse pressure increases as a direct result of the 1.4-percent power uprate. The turbine impulse pressure transmitter provides a percent turbine power load signal to the reference T_{avg} program for rod control system. Turbine impulse pressure also provides an input (P-13) to the RPS permissive interlock, P-7. The turbine impulse pressure will be re-scaled for the increase pressure at full power, to provide the desired full power reference T_{avg} at the uprated power.

Based on revised heat balances provided by the TG manufacturer, the uprating will result in operation at generator gross output powers of approximately 898 MWe (Unit 1) and 908 MWe (Unit 2). The increased power output is well within the generator nameplate rating of 1,026 Mva @0.9 PF. Therefore, the generator is capable of operating at the uprated power level with no modifications.

3.9 ELECTRICAL SYSTEMS

3.9.1 AC and DC Plant Electrical Systems

The electrical distribution systems were reviewed to identify the major items that may be affected by uprate conditions and to evaluate the potential impact of an uprate on that equipment. The following systems and components have been evaluated:

- Main unit generator
- Isolated phase bus ducts
- Main power transformers (MPTs)

- Unit station service transformers (USSTs)
- System station service transformers (SSSTs)
- Non-segregated phase bus ducts
- Large loads and cables
- Emergency diesel generators
- Protective relay settings

The "main generator" source and "switchyard" source provide at each unit the normal source and alternate source of power for all safety-related and nonsafety-related loads, respectively.

System reviews confirmed that only large, nonsafety-related, ac-powered loads were affected by unit operation at core uprate conditions. Additionally, the reviews confirmed that control of the affected loads remained unchanged. Accordingly, the direct current systems at each unit are unaffected by unit operation at uprate conditions.

At the current NSSS thermal rating, main generator gross output is approximately 893 MW and 888 MW at Unit 1 and Unit 2, respectively. The nameplate ratings for the main generator at each unit are: 1,026 Mva at 22 kV, 0.9 power factor with hydrogen at 75 psig. A review of the applicable generator reactive capability curve confirms that each machine is capable of operating at a maximum real power output of 1,026 MW at a 1.0 power factor (zero megavar output). Heat balance studies completed for a 1.4-percent uprate identify gross generator output levels less than the maximum. Machine operation at a lower real output power level and a power factor of 1.0, or less, is permissible provided unit operation remains within the real and reactive power limits defined by the reactive capability curve.

The isolated phase duct, main power transformers, and associated cooling equipment are designed to accept the maximum generator output and therefore will continue to support plant operations at uprate.

The bus loading summaries for connected 4,160V switchgears under uprate conditions remain less than the USSTs and SSSTs design ratings. The USST's associated cooling equipment will also support power uprate for continuous operation with no modifications.

The non-segregated phase duct connect the USSTs and SSSTs to their respective 4,160V switchgears. The non-segregated phase bus duct runs have a continuous rating of 2,500 Amps per phase at 4,160V. The bus loading summaries for connected 4,160V switchgears under uprate conditions confirm that the non-segregated phase ducts are adequate.

System evaluations determined that a few of the large medium voltage motors on nonsafety-related 4,160V switchgears experience a slight brake horsepower (BHP)/kW load change at power uprate, from present loading requirements. Load flow analysis performed for 4,160V bus loads under uprate conditions verify acceptable loading. Therefore, the large station auxiliary loads and associated cables are considered adequate as installed, and the motors will continue to satisfactorily perform their intended functions.

The ESF (safety-related) motors do not experience a BHP/kW load change at uprate conditions. Therefore, the diesel generators will not be impacted by power uprate and will remain capable of performing their safety-related functions during a LOOP/LOCA and power uprate.

All other electrical equipment and components, including station protective schemes and setpoints, will continue to support safe and reliable plant operation at uprate. Bus voltage and fault current values at different levels of the station auxiliary electrical distribution systems will remain within acceptable limits under uprate. In addition, there are no impacts to the DC power system voltage or short circuit current levels.

The switchyard equipment (345 kV switches and breakers) are rated 2,000 Amps, which exceeds at power uprate the main generator output current of approximately 1,700 Amps at its nameplate rating of 1,026 Mva and 345 kV. Therefore, the switchyard will accept the additional load without the need of any hardware modification.

The Beaver Valley Units 1 and 2 receive shutdown power from two physically independent and redundant offsite power sources of the 138 kV switchyard system. Under power uprate, there is no change in the shutdown (ESF) loads, and bus voltage values at different levels of the station auxiliary distribution systems are bounded by the existing load flow and voltage profile analysis. The 1.4-percent increase in power generated into the 345 kV has no significant impact on the 138 kV switchyard system and the ability of the units to safety shut down.

3.9.2 Grid Stability

The Beaver Valley Units 1 and 2 receive shutdown power from two physically independent and redundant offsite power sources of the 138 kV switchyard system. Under power uprate, there is no change in the shutdown (EFS) loads, and bus voltage values at different levels of the station auxiliary distribution systems are bounded by the existing load flow and voltage profile analysis. The 1.4-percent increase in power generated into the 345 kV has no significant impact on the 138 kV switchyard system and the ability of the units to safely shut down.

An Office of Nuclear Regulatory Research report, "The Effects of Deregulation of the Electric power Industry on the Nuclear Plant Offsite Power System: An Evaluation" dated June 30, 1999, recommended that grid stability analyses be updated by licensees periodically to reflect changes in the grid power system.

A grid stability study for Beaver Valley Station was performed in 1996 and 1997. The grid stability study indicated that the transmission system remains stable under worst case postulated contingencies, and that the calculated voltages at Beaver Valley 345 kV and 138 kV buses are acceptable.

Another update of the grid stability study is being performed at this time, to update the model with system changes that have occurred since 1997. The new study will incorporate the

1.4-percent power uprate to determine if any stability issues require resolution. This new study will be completed prior to increasing power above 2,652 MWt.

3.10 NUCLEAR STEAM SUPPLY SYSTEM ACCIDENT EVALUATION

3.10.1 Steam Generator Tube Rupture Evaluation

The licensing basis steam generator tube rupture (SGTR) analyses for Beaver Valley Units 1 and 2 are presented in their respective UFSARs. The SGTR analyses consist of a thermal and hydraulic analysis to determine the primary-to-secondary break flow and the steam released to the atmosphere, and a radiological consequences analysis to calculate the offsite radiation doses resulting from the event. The SGTR thermal and hydraulic analyses calculate the primary-to-secondary break flow and steam released to the atmosphere from the ruptured and intact steam generators for the time period before break flow termination. The analyses also calculate the long-term releases to the atmosphere from the intact steam generators after break flow termination. These results are then used to evaluate the offsite radiological consequences for an SGTR.

Unit 1 Evaluation

The current licensing basis SGTR thermal and hydraulic analysis for Unit 1 was performed using a simplified mass and energy balance method. The input parameters in the thermal and hydraulic analysis that are changing as a result of the power uprate are NSSS design parameters. These parameters include power, hot leg temperature, cold leg temperature, steam temperature, and steam pressure. However, the steam conditions in the steam generators are limited to minimums of 760 psia and 512.3°F due to component design transient considerations. Since the steam conditions are limited to minimum values, steam temperature and pressure considered in the analysis are not changing. Therefore, primary-to-secondary break flow is not affected. An increase in power and hot leg temperature results in an increase in steam release due to an increase in the system energy. The methodology used in the current licensing basis analysis includes a 4.5-percent margin in reactor power in the calculation of the feedwater flows and the steam releases. This 4.5-percent margin will cover the small increase in steam release due to the 1.4-percent power increase and minor changes to the design parameters. Since the steam release and the break flow determined in the current licensing basis analysis remain bounding, the input to the radiological consequences analysis is not affected by the power uprate.

Unit 2 Evaluation

The current licensing basis SGTR thermal and hydraulic analysis for Unit 2 was performed using the LOFTTR2 methodology. The LOFTTR2 SGTR analysis includes an analysis to demonstrate margin to steam generator overfill and a thermal and hydraulic analysis. The licensing basis analysis for the SGTR margin to overfill event is initiated from full power. For the current analysis, a 2-percent power measurement uncertainty is considered in the calculation, thereby increasing the initial power level to 102 percent. As the current licensing basis analysis was evaluated to 102 percent, an evaluation at 101.4-percent power is not required. Since the steam release and the break flow determined in the current licensing basis remain bounding, the input to the radiological consequences analysis is not affected by the power uprate.

3.10.2 LOCA-Related Analyses

3.10.2.1 LBLOCA and SBLOCA

The current licensing large-break LOCA (LBLOCA) and small-break LOCA (SBLOCA) analyses employ a nominal core power of 2,652 MWt. The licensing basis analysis methodology employs a 2-percent calorimetric uncertainty (yielding an assumed core power of 2,705 MWt) in accordance with the original requirements of 10CFR50, Appendix K. Consistent with the recent change to Appendix K, a reduction in power measurement uncertainty to 0.6 percent, based on the use of the Caldon LEFMs, is proposed. The existing 2-percent uncertainty margin in the LBLOCA and SBLOCA analyses would be re-allocated with 1.4 percent applied to the increase in licensed core power level and 0.6 percent retained to account for power measurement uncertainty.

3.10.2.2 Post-LOCA Long-Term Core Cooling (LTCC)

The licensing position for satisfying the requirements of 10CFR50.46, Paragraph (b), Item (5), "Long-term cooling," concludes that the reactor will remain shut down by borated emergency core cooling system (ECCS) water residing in the RCS/sump following a LOCA. Since credit for the control rods is not currently taken for an LBLOCA, the borated ECCS water provided by the RWST and accumulators must have a boron concentration that, when mixed with other sources of water, will result in the reactor core remaining subcritical assuming all control rods out. The calculation is based upon the reactor steady-state conditions at the initiation of a LOCA and considers sources of both borated and unborated fluid in the post-LOCA containment sump. The other sources of water considered in the calculation of the sump boron concentration are the RCS, ECCS/residual heat removal piping, and the chemical addition tank. An evaluation of required ECCS water volumes and boric acid concentrations will be performed as part of the normal RSE process.

3.10.2.3 Hot Leg Switchover

For a cold leg break post-LOCA, some of the ECCS injection into the cold leg will circulate around the top of the full downcomer and out of the broken cold leg. Flow stagnation in the core and the boiling off of near pure water will increase the boron concentration of the remaining water. As the boron concentration increases, the boron will eventually precipitate and potentially inhibit core cooling. Therefore, at a designated time after a LOCA, the ECCS configuration is switched to hot leg injection to flush the core with water and keep the boron concentration below the precipitation point. The licensing basis analysis methodology employs a 2-percent calorimetric uncertainty in accordance with the original requirements of 10CFR50, Appendix K. Consistent with the recent change to Appendix K, a reduction in power measurement uncertainty to 0.6 percent, based on the use of the LEFMs, is proposed. The existing 2-percent uncertainty margin in hot leg switchover analysis would be re-allocated with

1.4-percent applied to the increase in licensed power level and 0.6 percent retained to account for power measurement uncertainty.

3.10.3 Reactor Vessel, Loop, and Steam Generator LOCA Forces Evaluation

The purpose of a LOCA hydraulic forces analysis is to generate the hydraulic forcing functions and hydraulic loads that occur on RCS components as a result of a postulated LOCA. These forcing functions and loads are considered in the structural design of the NSSS components.

In support of the 1.4-percent uprating conditions for Beaver Valley Units 1 and 2, an evaluation of the impact of uprated RCS conditions from Section 3.3 on the LOCA forces was performed. This evaluation demonstrates that the LOCA forces Analyses of Record are bounding for the 1.4-percent uprate program.

Unit 1 Evaluation

The vessel LOCA forces evaluation applied margin gained from reducing the reactor vessel break area to that of the limiting branch line break. This is allowed given previous approval of LBB methodology to the main coolant loop piping breaks. The estimated increase to the LOCA forces due to the change in RCS temperatures for the uprate was then compared to the estimated decrease in LOCA forces due to the break area reduction. The effect of the break area reduction more than offsets the increase, resulting in vessel LOCA forces conservatively estimated as having more than 4.5-percent margin remaining.

The loop and steam generator LOCA forces evaluation applied margin gained from employing a more accurate model of the loop at the break location. The original analyses conservatively applied a branch line nozzle flow area equivalent to the main coolant loop piping. The approved methodology for modeling branch line breaks using MULTIFLEX (as documented in WCAP-8082-P-A, "Pipe Breaks for the LOCA Analysis of the Westinghouse Primary Coolant Loop") indicates that branch line breaks can be postulated at the safe end of the nozzle. Sensitivities to changing the branch line area are used to conservatively estimate the reduction in peak force at the break location. The net effect of the uprating and the more accurate model of the loop at the break location results in an approximately 4-percent margin remaining in the loop LOCA forces, and approximately 14-percent margin remaining in the steam generator LOCA forces.

Unit 2 Evaluation

Break area reduction margin is used, as allowed with LBB methodology, for all lines of 6-inch nominal size or larger. The estimated increase in LOCA hydraulic forces due to the change in RCS temperatures for the uprate was then compared to the estimated decrease in LOCA forces due to the break area reduction. The comparison shows that the LOCA force reduction from the break area margin more than offsets the increase in LOCA forces associated with the uprate conditions. Specifically, more than 6-percent margin remains for the vessel LOCA forces, and approximately 14-percent margin remains in both the loop LOCA forces and the steam generator LOCA forces.

Conclusion

It is concluded that the existing LOCA hydraulic forces Analyses of Record supporting Beaver Valley Units 1 and 2 remain conservative at the 1.4-percent uprated conditions.

3.10.4 Non-LOCA/Transient Analyses

Unit 1 Evaluations

All of the UFSAR Chapter 14 non-LOCA analyses applicable to Beaver Valley Unit 1 were reviewed to determine their continued acceptability for operation considering the 1.4-percent power uprate conditions. The evaluation of these non-LOCA analyses was performed concurrently with the recent assessment of the change to the Revised Thermal Design Procedure (RTDP) for Unit 1. The evaluations and results are presented in the RTDP Analysis Report for Unit 1. All applicable acceptance criteria for each of the analyzed events continue to be met.

Unit 2 Evaluations

All of the UFSAR Chapter 15 non-LOCA analyses applicable to Beaver Valley Unit 2 were reviewed to determine their continued acceptability for operation considering the 1.4-percent power uprate conditions. The evaluation of these non-LOCA analyses was performed concurrently with the recent assessment of the change to the RTDP for Unit 2. The evaluations and results are presented in the RTDP Analysis Report for Unit 2 with Addendum. All applicable acceptance criteria for each of the analyzed events continue to be met.

Conclusion

All applicable acceptance criteria for the non-LOCA events continue to be met for both Beaver Valley Units 1 and 2.

3.10.5 Revised Thermal Design Procedures Uncertainties

It should be noted that separate amendments, LAR-286 for Unit 1 and LAR-158 for Unit 2, have been submitted that implement the use of Revised Thermal Design Procedure methodology. Westinghouse WCAP-15264 Revision 3 ("Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology for FirstEnergy Nuclear Operating Company Beaver Valley Unit 1") and WCAP-15265, Revision 2 ("Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology for FirstEnergy Nuclear Operating Company Beaver Valley Unit 2") provide the basis for the RTDP uncertainties that are used in the Beaver Valley safety analyses. These include power measurement (calorimetric), T_{avg} (rod) control, pressurizer pressure control, and RCS flow measurement (calorimetric) and indication. The effect of the power uprating on these uncertainties is discussed in the following subsections.

3.10.5.1 Power Calorimetric

Typical plant safety analysis evaluations assume a power calorimetric uncertainty of 2.0-percent RTP. This power uprate concept is based on a reduction in the power calorimetric uncertainties such that the calculated uncertainties plus the magnitude of the power uprate remains within the original plant analysis assumptions. Therefore, the final calculated uncertainties are used to determine the magnitude of the power uprate. The primary source in reducing the power calorimetric uncertainties is a reduction in the uncertainties associated with the measurement of secondary-side feedwater mass flow. New calculations were performed to determine the uncertainties for the daily power calorimetric assuming the use of the Caldon LEFM (Unit 1) and the Caldon CheckPlus LEFM (Unit 2) measurement systems to determine total feedwater mass flow. The mass flow error, in combination with the remaining uncertainty components, results in a total 95/95 power measurement uncertainty of ±0.6-percent RTP. A power measurement uncertainty of ±0.6 percent justifies a power uprate of 1.4-percent RTP.

Thermal power measurements are used to calibrate the Nuclear Instrumentation System (NIS) excore detectors, which input to the Reactor Protection System to protect the core from overpower events. The reactor trip system setpoints that currently utilize NIS excore signals with an uncertainty allowance for the power calorimetric of ±2% RTP will conservatively continue to retain this ±2 allowance.

3.10.5.2 T_{avg} (Rod) Control, Pressurizer Pressure Control and RCS Flow Calorimetric

The uncertainties associated with the pressurizer pressure control system are not affected by changes in plant parameters for the 1.4-percent power uprate conditions. T_{avg} (rod) control assumes the nominal full-power turbine impulse pressure for the T_{avg} reference. The minor change in full-power turbine impulse pressure does not affect the final calculated T_{avg} (rod) control uncertainties. Therefore, the uprate does not necessitate changes to the uncertainties documented for these controllers. For the RCS flow calorimetric, the small changes in the plant parameters due to the uprate conditions do not change the final calculated uncertainties as documented in WCAP-15264, Revision 3 (Unit 1) and WCAP-15265, Revision 2 (Unit 2).

3.10.5.3 RTS/ESFAS Uncertainties

Westinghouse WCAP-11419, Revision 2 ("Westinghouse Setpoint Methodology for Protection Systems Beaver Valley Unit 1") and WCAP-11366, Revision 4 ("Westinghouse Setpoint Methodology for Protection Systems Beaver Valley Unit 2") provide the basis for the reactor trip system (RTS)/ESFAS uncertainties that are used in the Beaver Valley safety analyses. These include loss of flow, steam generator water level, overtemperature (OTΔT), and overpower ΔT (OPΔT). The effect of the power uprating on these uncertainties is discussed in the following subsections. All the other RTS/ESFAS functions are unaffected by the uprating.

3.10.5.3.1 RCS Loss of Flow

The small changes in the plant parameters due to the power uprate conditions do not change the final calculated RCS flow calorimetric uncertainties. Therefore, the uprate does not necessitate changes to the uncertainties documented in WCAP-11419, Revision 2 (Unit 1) and WCAP-11366, Revision 4 (Unit 2) for the RCS loss of flow trip.

3.10.5.3.2 Steam Generator Water Level

The small change in nominal steam pressure and feedwater temperature due to the power uprate conditions does not change the final calculated steam generator water level channel uncertainties. Other potential contributors to level measurement uncertainty, including recirculation ratio, reference leg temperature effects, were found to be not significantly affected by the proposed uprating. Therefore, the uprate does not necessitate changes to the uncertainties documented in WCAP-11419, Revision 2 (Unit 1) and WCAP-11366, Revision 4 (Unit 2) for the steam generator water level trip(s).

3.10.5.3.3 Over-Temperature ΔT and Over-Power ΔT

Full-power ΔT will increase due to the power uprate conditions, which is conservative for the calculated uncertainties. Therefore, the uprate does not necessitate changes to the uncertainties documented in WCAP-11419, Revision 2 (Unit 1) and WCAP-11366, Revision 4 (Unit 2) for the OTΔT and OPΔT trips.

3.11 CONTAINMENT/BOP ACCIDENT EVALUATIONS

3.11.1 Mass and Energy Release Data

3.11.1.1 Short-Term LOCA Mass and Energy Release Data

Containment analyses demonstrate the adequacy of the Containment Building and its internal walls, and qualify the equipment inside containment for a LOCA. A short-term LOCA analysis was performed to determine compartment pressurization of subcompartments located inside containment. This section discusses the impact of the 1.4-percent uprate on the short-term LOCA mass and energy analyses.

Short-term LOCA mass and energy release calculations were performed to support the reactor cavity and loop subcompartment pressurization analyses. These analyses were performed to ensure that the walls in the immediate proximity of the break location can maintain their structural integrity during the short pressure pulse that accompanies a LOCA within the region.

Per the Beaver Valley UFSAR, subcompartment analyses were conducted for the upper pressurizer cubicle (spray line break), lower pressurizer cubicle (surge line break), pressurizer relief tank cubicle (surge line break), steam generator subcompartments (RCS split break at the steam generator inlet elbow), and the reactor cavity (150 in^2 RCS cold leg break).

The approved methodology for the Unit 2 licensing basis analysis is documented in WCAP-8264-P-A, Revision 1, "Topical Report Westinghouse Mass and Energy Release Data for Containment Design" The critical flow calculation employs appropriately defined critical flow correlations applied for fluid conditions at the break location. For the early portion of blowdown, subcooled, saturated, and two-phase critical flow regions are encountered. The methodology of WCAP-8264-P-A uses the Moody correlation for saturated and two-phase conditions and a modification of the Zaloudek correlation for the subcooled blowdown regime. The details of these models and comparisons to other models and experimental data are described in the WCAP. Most short-term blowdown transients are characterized by a peak mass and energy release rate that occurs during a subcooled condition. The Zaloudek correlation, which models this condition, is currently used in the short-term LOCA mass and energy release analyses. The analysis input that may potentially change with the uprate is the initial RCS fluid temperature. The use of lower temperatures maximizes the critical mass flux in the Zaloudek correlation. Since this event lasts for approximately 3 seconds, the single effect of power is not significant.

Based upon acceptance of LBB, the dynamic effects associated with RCS pipe breaks are no longer considered within the licensing basis. The releases and compartment analyses associated with these breaks are conservative with respect to all remaining credible breaks. The benefits of the decrease in mass and energy releases associated with the smaller RCS nozzle breaks, as compared to the larger RCS pipe breaks, more than offset any penalties associated with possible increased releases that will result from decreased RCS cold leg temperatures. Therefore, the design of the reactor loop compartments and reactor cavity region remains acceptable at the uprated power level.

The pressurizer surge line is also no longer considered in the licensing basis due to LBB. Therefore, the lower and middle pressurizer compartments remain unaffected.

For the upper pressurizer cubicle design, the spray line is the limiting break, and is not eliminated with LBB. The current Unit 2 releases, which are based upon a conservative break flow area, remain bounding for the 1.4-percent uprate. Consideration of the break flow area based upon the actual pipe size more than offsets the possible increase in releases due to the uprate. Therefore, the current mass and energy releases for the spray line remain bounding. For Unit 1, an evaluation has been completed that demonstrates sufficient margin exists within the compartment structural design to accommodate the small increase in releases due to the uprate.

3.11.1.2 Long-Term LOCA Mass and Energy Release Data

The current Unit 1 mass and energy release data are based on a power level of 2,713 MWt, which is bounding for the power uprate.

The current Unit 2 pressure analysis uses a combination of NSSS power levels. In the early stages of the accident, the mass and energy release data are based on a power level of 2,811 MWt. In the latter portion of the analysis, they are based on a power level of 2,713 MWt. Both of these power levels bound the power uprate.

3.11.1.3 Main Steam Line Mass and Energy Release Data

The licensing basis safety analyses related to steam line break mass and energy releases were evaluated to determine the effect of a power uprate of up to 1.4 percent for Beaver Valley Units 1 and 2.

3.11.1.3.1 Steam Line Break Mass and Energy Releases Inside and Outside Containment

Critical parameters for the steam line break event affected by the 1.4-percent power uprate include the following conditions:

- NSSS power level
- Main feedwater flow and temperature

The power increase of 1.4 percent for the two Beaver Valley units will be offset by an equivalent reduction in the calorimetric uncertainty. The Analyses of Record applicable to both units for the inside and outside containment long-term steam line breaks assume a 2-percent power calorimetric uncertainty on a 2,660 MWt NSSS power (2,652 MWt core power). A maximum 0.6-percent power calorimetric uncertainty applied to a 1.4-percent power increase is equivalent to the licensing basis safety analysis for Beaver Valley.

There is a small increase in the feedwater flow and feedwater temperatures for full-power operation. At lower power, the parameter differences would be minimal. This small increase has a minimal effect on the transient mass and energy releases inside containment.

For the steam line break releases outside containment, the expected effect on the peak compartments ambient temperature response due to the secondary-side flow and temperature changes is an increase of 2°F. This is discussed further in Section 3.11.3.

3.11.2 Containment Analysis

3.11.2.1 MSLB and LOCA

As stated in Section 3.11.1.2, the mass and energy release data for the LOCA bound the power uprate conditions. Therefore, the peak LOCA containment pressure and temperature will not be impacted by the power uprate. The spray system capability to return the containment to sub-atmospheric pressure conditions following a LOCA is also not impacted by the power uprate.

Section 3.11.1.3 indicates that the main steam line break changes in the mass and energy release data will not be significant. Therefore, the resultant pressures and temperature will not be significantly affected.

3.11.2.2 Combustible Gas Control

An increase in the power level will increase the hydrogen generation rate due to radiolytic decomposition of water. This mechanism is responsible for a majority of the hydrogen generation. Since the current analyses for Units 1 and 2 uses a core power level of 2,705 MWt, the results bound the core power uprate to 2,689 MWt.

3.11.3 Equipment Qualification Accident Environments

The analysis of accident environments for equipment qualification is evaluated in two parts: LOCA and main steam line break inside containment; and high-energy lines outside containment.

3.11.3.1 LOCA and Main Steam Line Break Inside Containment

As stated in Section 3.11.2, the current containment LOCA and main steam line break analyses will not be affected by uprate conditions. The current equipment qualification accident environments inside containment bound the environments resulting from the power uprate.

3.11.3.2 High-Energy Line Breaks Outside Containment

For a steam line break outside containment, the expected effect on the peak compartment ambient temperature response is an increase of 2° to 3°F. Margin exists in the Unit 1 temperature envelope to bound the small increase in temperature for the uprate conditions. The Unit 2 equipment is evaluated via thermal lag analysis driven by the ambient temperature transient, the 2° to 3°F temperature increase is insignificant with respect to environmental qualification of the equipment.

3.12 RADIOLOGICAL CONSEQUENCES

The current licensed NSSS power level for both Units 1 and 2 is 2,660 MWt, which is also the warranted NSSS power output. Since 8 MWt is developed by the reactor coolant pumps, the licensed core level is 2,652 MWt. In accordance with the guidance of Regulatory Guide 1.49 to address possible instrument error in determining the power level, post-accident radiological analyses performed to demonstrate compliance with 10CFR100 should be based on at least 1.02 times the proposed licensed core level. This resulted in the need to use a core power level of 2,705 MWt or greater in the radiological accident analyses for Units 1 and 2.

The radiological analyses currently supporting Units 1 and 2 for normal operation are based on a power level 2,766 MWt with a 12-month operating cycle (i.e., approximately 4 percent above the licensed core power level). The radiological accident analyses for Units 1 and 2 are based on a power level of 2,705 MWt, which is consistent with the current design basis.

3.12.1 Normal Operation Analyses

3.12.1.1 Radiation Source Terms

The NSSS power uprate to 2,697 MWt will increase the concentration of fission products in the primary and secondary coolant by approximately 1.4 percent. The expected source terms, which are generated based on the power level, will increase by approximately the same percentage. However, the current design basis radiation source terms are based on a power level of 2,766 MWt, with a 12-month operating cycle. In addition, the technical specification source terms will not change since:

- The power uprate increases the concentration of tritium and fission products in the coolant, but does not significantly impact the mix of radionuclides.

- The source terms are calculated based on normalization to the I-131 dose equivalent, which is fixed via the Technical Specifications.

3.12.1.2 Gaseous and Liquid Releases

The Unit 1 and 2 gaseous and liquid effluent releases reported in the Beaver Valley Unit 2 UFSAR, as well as the Appendix I analyses, are based on both units operating at power levels of 2,766 MWt and therefore bound the power uprate conditions.

Release concentrations and offsite doses are controlled by the BV-1 and BV-2 Offsite Dose Calculation Manual (ODCM). This manual provides the information and methodologies to be used at Beaver Valley Units 1 and 2 to assure compliance with the Administrative Controls Section (i.e., 6.8.6a) of the Technical Specifications. Specifically, Appendix C of the ODCM contains controls necessary to show compliance with 10CFR20.1302, 40CFR Part 190, 10CFR50.36a, and Appendix I to 10CFR50. Compliance with these controls further ensures both the accuracy and reliability of effluent dose calculations, and effluent alarm setpoint calculations.

3.12.1.3 Shielding

The Units 1 and 2 cubicle gamma and neutron shielding designs are based on 1-percent failed fuel and a power of 2,766 MWt.

3.12.1.4 Normal Operation Analyses - Summary

Based on the discussions provided above, an NSSS power uprate to 2,697 MWt will not cause radiological exposure in excess of the dose criteria (for restricted and unrestricted access) provided in the current 10CFR20. From an operations perspective, radiation levels in most areas of the plant are expected to increase no more than the percentage increase in power level. Individual worker exposures will be maintained within acceptable limits by the site ALARA Program, which controls access to radiation areas. Gaseous and Liquid Effluent releases are also expected to increase by no more than the percentage increase in power level. Offsite release concentrations and doses will be maintained within the limits of the current 10CFR20 and 10CFR50, Appendix I by the site Radwaste Effluent Control Program.

3.12.2 Accident Analyses

Beaver Valley Units 1 and 2 have submitted to the NRC licensing amendments LAR-280 (Unit 1) and LAR-151 (Unit 2) that (Ref. FirstEnergy Operating Company letter L-00-008, dated May 12, 2000) that modified information on the design basis accident radiological dose analyses presented in the UFSAR. The revised design basis accident radiological dose analyses were performed based on a reactor power of 2,705 MWt for 18 month cycles. They demonstrate that the dose limits set by 10CFR100 and 10CFR50, Appendix A, General Design Criterion 19 for the site boundary and control room, respectively, are met.

3.12.3 Equipment Qualification

The qualified life of Class-1E and other post-accident monitoring (PAM) instrumentation has been reviewed for impact due to the environmental conditions resulting from operating the plant at uprated power. The environmental radiation levels for both normal operation and accident conditions were originally developed using assumed power levels that envelope the uprated condition.

For the accident contribution, margins were incorporated into the equipment specifications that met or exceeded the requirements of IEEE-323-1974. In addition, relative to normal operation conditions, the actual exposures in the plant historically have been substantially below the assumptions used in the EQ dose calculations, so additional margin is being created as time passes.

Therefore on the basis of these considerations, it is acceptable to operate at the uprated power.

3.13 NUCLEAR FUEL

This section summarizes the evaluations performed to determine the effect of the 1.4-percent uprating on the nuclear fuel. The core design for Beaver Valley Units 1 and 2 is performed for each specific fuel cycle and varies according to the needs and specifications for each cycle. However, some fuel-related analyses are not cycle specific. The nuclear fuel review for the 1.4-percent uprate evaluated the fuel core design, thermal-hydraulic design, and fuel rod performance.

3.13.1 Fuel Core Design

A representative equilibrium cycle model was developed to evaluate the effects of the 1.4-percent uprate conditions on the fuel core design. Since the power uprate is relatively small, the representative cycle is adequate to demonstrate the sensitivity of reload parameters to the power uprate conditions. The expected ranges of variation in key parameters were determined.

The methods and core models used in the uprate analyses are consistent with those presented in the Beaver Valley units UFSARs. No changes to the nuclear design philosophy, methods, or models are necessary due to the uprating. The core analyses for the uprating were performed primarily to determine if the values previously used for the key safety parameters remain applicable prior to the cycle-specific reload design.

The core analyses show that the implementation of the power uprate does not result in changes to the current nuclear design basis documented in the UFSAR. The impact of the uprate on peaking factors, rod worths, reactivity coefficients, shutdown margin, and kinetics parameters is either well within normal cycle-to-cycle variation of these values or controlled by the core design and will be addressed on a cycle-specific basis consistent with reload methodology.

3.13.2 Core Thermal-Hydraulic Design

The core thermal-hydraulic analyses and evaluations were performed at the uprated core power level of 2689 MWt. The analyses assumed that the uprated core designs are composed of V5H fuel assemblies without Intermediate Flow Mixers (IFMs).

The thermal-hydraulic design method and computer code used for the 1.4-percent uprating to meet the DNB design basis are the RTDP and the THINC IV code. The WRB-1 DNB correlation is used for the 17x17 V5H fuel assemblies.

3.13.3 Fuel Rod Design

The fuel rod design analyses of Beaver Valley Unit 1 (Cycle 14) and Unit 2 (Cycle 9) were reviewed to assess the impact of a 1.4-percent power uprate. The rod internal pressure (gap reopening) and cladding stress, the two most impacted criteria, were re-evaluated under uprated conditions. The results show they continue to meet the acceptance criteria with significant margin. The other fuel rod design criteria are negligibly impacted by an increase of the power level, and sufficient margin currently exists to offset the result of the 1.4-percent power uprate.

4.0 PROGRAMS

The power uprate has the potential to affect programs that are developed and implemented by station personnel to demonstrate that topical areas comply with various design and licensing requirements. The plant programs and/or issues listed in Table 4-1 were reviewed to determine the impact due to the power uprate. In addition to the programs, plant Technical Specifications address specific requirements for a number of programs. These programs are identified in Table 4-2.

For the programs listed in Table 4-1, the controlling procedures and processes for the programs and key reference items within the procedures were reviewed. Program sponsors implementing organization personnel and other cognizant individuals were interviewed for those issues and programs that would be impacted by the uprate. Based upon the review of this information, the extent of impact by the implementation of the power uprate was determined for the various issues and programs.

For the programs listed in Table 4-2, the Technical Specifications and License Requirements Manual Sections associated with the programs were reviewed to identify any areas affected by power uprate.

The review process resulted in two groupings; not affected; and affected but changes would be captured by in-place processes and procedures that the power uprate information would be incorporated into the affected programs. The results of the review identified two programs that would be impacted by the uprate. However, changes to these programs will be captured by in-place change procedures as identified below:

Simulator

The Beaver Valley Unit specific simulator, which mimics the actual control room and is primarily used for training of operations personnel. In addition to the overall physical likeness between the actual control room and the simulator, computer systems provide simulator responses that are intended to match actual plant conditions for the simulation of accidents and transients, to the greatest extent possible. To ensure that the simulator accurately reflects the plant status, physical appearance (hardware) and simulation of plant response (software), changes resulting from the power uprate must be effectively communicated.

A review of the training simulator fidelity with the new power rating will be included at the next regularly scheduled review following the uprating in RTP. Simulator revalidation is performed in accordance with ANSI/ANS 3.5-1985.

Physical changes (hardware) that affect the control room and the simulator will be implemented through plant approved change processes. Copies of these change processes are procedurally routed to the Training Department and the simulator personnel implement appropriate changes.

Included with the design modification package for the uprating will be implementation of all the necessary procedures and training documents required for operation at the uprated power level with the new LEFM System.

The implementation of the power uprate will also result in changes in plant operating characteristics (software changes). These changes will range from simple changes in process temperatures and flow rates to plant responses to accidents and transients.

Flow Accelerated Corrosion (FAC)

The main feedwater systems, as well as other power conversion systems, are important to safe operation. Failures of passive components in these systems, such as piping can result in undesirable challenges to plant safety systems required for safe shutdown and accident mitigation. Failure of high-energy piping, such as feedwater system piping, can result in complex challenges to operating staff and the plant because of potential system interactions of high-energy steam and water with other systems, such as electrical distribution, fire protection, and security. FENOC has committed to adhere to criteria, codes and standards for high-energy piping systems described in licensing documents. Such commitments are a part of the licensing basis for the facility. An important part of this commitment is that piping will be maintained within allowable thickness values.

FAC, in the piping systems at Beaver Valley, is modeled using the CHECWORKS computer program. CHECWORKS models will be revised, as appropriate, to incorporate flow and thermodynamic states that are projected for uprated conditions. The results of these models will be factored into future inspection/pipe replacement plans consistent with the current FAC Program requirements.

4.1 RESPONSE TO PREVIOUS NRC UPRATE RAI ON OPERATING PROCEDURES (ABNORMAL/NORMAL) AND OPERATOR ACTIONS

The 1.4-percent power uprate is not expected to have any significant effect on the manner in which the operators control the plant, either during normal operations or transient conditions. The power uprate will lead to minor changes in several plant parameters. These parameters may include, but are not limited to, the 100-percent value for rated thermal power, reactor coolant system delta temperature, main turbine impulse pressure, steam generator pressure, and main feedwater and steam flows. Changes associated with the power uprate will be treated in a manner consistent with any other plant modification, and will be included in operator training accordingly.

4.2 RESPONSE TO PREVIOUS NRC UPRATE RAI ON STATION BLACKOUT EVENT

An evaluation of the SBO Analysis was performed to determine the impact of the power uprate. The results of this evaluation demonstrated that there was no impact on the ability to achieve and maintain safety shutdown of one unit at Beaver Valley Power Station during and following an SBO event. The evaluation included: heat-up analysis, equipment operability and battery capacity.

4.3 RESPONSE TO PREVIOUS NRC UPRATE RAI ON GENERIC LETTERS 89-10, 95-07 AND 96-06

4.3.1 Generic Letter 89-10 "Safety Related Motor-Operated Valve Testing and Surveillance"

As a result of the 1.4-percent power uprate, there are no required changes to the BVPS GL 89-10 MOV Program. Design basis differential pressures developed from conservative assumptions are used for MOV sizing requirements. These conditions bound uprate conditions and do not compromise margin of safety.

4.3.2 Generic Letter 95-07 "Pressure Locking and Thermal Binding of Safety Related Operated Gate Valves"

A review of the documentation and evaluations of GL 95-07 was performed to determine if the proposed 1.4-percent power increase would adversely affect any conclusions or qualifications that were approved by the NRC upon closure of the subject Generic Letter.

The conditions detailed in the evaluation remain bounding for the 1.4-percent power uprate. Conditions, conclusions and the bases for these conclusions as originally understood by the NRC, are unchanged and remain valid.

4.3.3 Generic Letter 96-06 "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions"

A review of the existing documentation and evaluations of GL 96-06 was performed to determine if the proposed 1.4-percent power increase would adversely affect any of the previous conclusions related to containment integrity (i.e., relative to overpressurization of safety related, water filled, isolable piping sections inside containment).

Conditions detailed in the evaluation remain bounding for the 1.4-percent power uprate. The post accident environments inside containment have not changed as a result of the power uprate. Therefore the isolable piping sections are not impacted.

4.4 RESPONSE TO PREVIOUS NRC UPRATE RAI ON ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)

In compliance with 10CFR50.62, ATWS mitigation system actuation circuitry (AMSAC) has been incorporated into the design of the BVPS units, based upon the recommendations of WCAP-10858P-A, Rev. 1 (Westinghouse, 1987).

AMSAC functions to protect the RCS from overpressure by starting the AFW pumps on loss of normal feedwater flow in any 2-out-of-3 loops. The system is armed when the C-20 permissive is met by operating above 40-percent power. The 1.4-percent power uprate will not affect the ability of AMSAC to perform its intended functions. The C-20 setpoint has been reviewed, and found to be acceptable without requiring modification due to the uprated power condition.

4.5 RESPONSE TO PREVIOUS NRC UPRATE RAI ON IPE

The Beaver Valley Probabilistic Risk Assessment (PRA) models include both a Level 1 Core Damage Frequency (CDF) analysis, and a Level 2 Large Early Release Frequency (LERF) analysis.

The success criteria for the Level 1 analyses were derived primarily from using UFSAR analyses, and as such were already analyzed using a 102-percent core power level (2,705 MWt). Since the proposed uprating is based on reducing the 2-percent margin for power measurement uncertainty that has been used typically in the UFSAR analyses, therefore remain bounded by the uprated power conditions.

Some success criteria derivations were performed using MAAP with a nominal core power level of 2,652 MWt. To determine the impact of the 1.4-percent core power uprate of these MAAP scenarios, some cases were rerun using the higher core power at 2,689 MWt. As expected there were some minor impacts to the timing of events (e.g., time to core uncovery). However, these were very minor in nature and are not expected to impact the success criteria of systems used to mitigate these analyzed conditions.

Likewise, the Level 2 analyses based on MAAP are only expected to have minor timing impacts from the 1.4-percent uprate on the Level 2 containment release analyses and not any significant changes on the release magnitudes. Therefore, the 1.4-percent core power uprate is not expected to have any significant impact on the PRA results.

4.6 RESPONSE TO PREVIOUS NRC UPRATE RAI ON ROD EJECTION EVENT

The fuel pellet enthalpy criterion is the same as that found in UFSAR Chapter 14.2.6 for Beaver Valley Unit 1 and in Chapter 15.4.8 for Beaver Valley Unit 2 (200 cal/gm for irradiated fuel and 225 cal/gm for unirradiated fuel.) In the Westinghouse methodology, a limit of 200 cal/gm is used since it bounds both irradiated and unirradiated fuel, as well as the Standard Review Plan value of 280 cal/gm.

The issues associated with the current Rod Ejection criterion and high burnup fuel were recently discussed in a meeting of the Westinghouse Fuel Working Group held on May 4, 2000, in Columbia S.C. NRC representatives also attended this meeting. In response to a question, an NRC representative indicated that there were no current plans to backfit any reduced fuel limits for the RCCA Ejection accident to plants that stay within the current generic licensed burnup limit (62,000 MWd/MTU). The intent is to apply these revised analysis limits, when available, only to plants requesting an increase in the licensed burnup limit. This position is consistent with the NRC's Memorandum "Agency Program Plan for High Burnup Fuel," from L. J. Callan to the ACRS, dated July 6, 1998. Current generic fuel license limits prohibit lead fuel rod burnups greater than 62,000 MWd/MTU. Therefore, the 200 cal/gm peak fuel enthalpy requirement applied by Westinghouse, which bounds the current 280 cal/gm criterion, is still applicable to Beaver Valley. Based upon the use of a 200 cal/gm limit, the Beaver Valley Rod Ejection analysis is consistent with section 15.4.8 of the Standard Review Plan and meets the requirements of General Design Criterion 28.

With respect to the reactor pressure criterion in Section 15.4.8 of the Standard Review Plan, Westinghouse has generically shown that this criterion is met as documented in WCAP-7588 Revision 1-A. Since this generic evaluation is applicable to Beaver Valley, the reactor pressure criterion is met for the Rod Ejection analysis.

The clad temperature criterion is the same as that found in the Beaver Valley UFSARs: average clad temperature at the hot spot below 3000°F. This is actually an internal criterion that was used by Westinghouse to provide an indication of core coolability and is not a licensing limit. As discussed in letter NS-NRC-89-3466, Westinghouse recognizes that the fuel pellet enthalpy limit and not clad temperature is the accepted criterion for confirming core coolability following the event. The criterion and discussion regarding the fuel cladding temperature limit was provided for informational purposes. The fuel enthalpy criterion, 200 cal/gm, continues to be used to demonstrate core coolability. It was determined that the enthalpy was less than 200 cal/gm for the 1.4% uprate conditions. Therefore, the core coolability would be expected to be maintained.

Table 4-1 Program/Issues

Issues and Programs	Requires Update
Plant Simulator	YES
Fire Protection (Appendix R)	NO
Check Valves	NO
Motor-Operated Valve Administrative Program (GL 89-10)	NO
Air-Operated Valves	NO
River/Service Water System Control and Monitoring (GL 89-13)	NO
Inservice Inspection Program	NO
Inservice Test Program	NO
Containment Integrity (Appendix J)	NO
Equipment Qualification	NO
Human Factors	NO
Station Blackout	NO
Anticipated Transient Without Scram	NO
Flow-Accelerated Corrosion Program	YES

No - Programs not impacted by uprate change or are bounded by existing analysis.
Yes - Programs impacted and changes to be addressed in uprate implementation.

Table 4-2 Technical Specification Programs

Program	Requires Update
Post-Accident Monitoring Program (TS 6.8.4)	NO
Secondary Water Chemistry Program (TS 6.8.5a)	NO
Radioactive Effluent Controls Program (TS 6.8.6a)	NO
Radioactive Environmental Monitoring Program (TS 6.8.6b)	NO
Radiation Protection Program (TS 6.11)	NO
Process Control Program (TS 6.13)	NO
Containment Leakage Rate Testing Program (TS 6.17)	NO

5.0 ENVIRONMENTAL IMPACT CONSIDERATION

The environmental review conducted for the proposed power uprate assessed the existing operating license and National Pollutant Discharge Elimination System (NPDES) permit limits and the information contained in the Final Environmental Report (FER). This assessment included determining whether the power uprate would cause the plant to exceed discharge limitations and NPDES permit conditions associated with the operation of the plant. In addition, a review of the recent Beaver Valley Units 1 and 2 Annual Radioactive Effluent Discharge Reports demonstrates that the actual releases from the plants are a very small percentage of the Technical Specification allowable limits and the FER estimates. The discharge amounts will not be significantly increased by the thermal power uprate and will continue to be a small percentage of the allowable limits and the FER estimates.

Onsite and offsite radiation exposures from normal operation and postulated accidents are addressed in Section 3.12. The offsite doses for the exposure postulated under accident conditions remain within the guidelines of the Code of Federal Regulations (CFR) 10CFR100.

The FER assessed the non-radiological impacts of plant operation as a function of plant design features, relative loss of renewable resources, and relative loss or degradation of available habitat. Environmental impacts associated with 40-year operating licenses were originally evaluated in the FER. After weighing the environmental, economic, technical, and other benefits against environmental costs and considering available alternatives, and subject to certain conditions, from the standpoint of environmental effects, the FER concluded that the issuance of operating licenses for Beaver Valley Units 1 and 2 was an acceptable action. These assessments, and the assumptions on which they were based, remain valid and are not impacted as a result of the thermal power uprate.

Protection of the environment is assured by compliance with permits issued by federal, state, and local agencies.

National Pollutant Discharge Elimination System Permit Impact

The Beaver Valley Plant consists of two nuclear units (Units 1 and 2). The two units employ a closed-loop cooling system that includes a natural draft cooling tower (one per unit) to dissipate waste heat to the atmosphere. All water used at the plant is recycled within the closed-loop cooling system except station makeup that comes from the Ohio River via the service water system. The cooling towers and circulating water system are addressed in Section 3.8.

The Beaver Valley NPDES permit (Permit No. PA0025615) does not place any operating limits on either flow or temperature.

The heat duty increase associated with uprate is mainly associated with the circulating water system and will be approximately 120 x 10^6 Btu/hr. This represents a 1.4-percent increase over the present power level, but is insignificant when compared to the current heat load from the two units. The maximum circulating water temperature increase expected as a result of uprate will be approximately 0.5°F over existing plant operation. Therefore, the thermal power uprate of the Beaver Valley Units 1 and 2 will have no adverse impacts on the environment or result in the exceeding NPDES permit limits.

Environmental Impact Consideration Summary

The proposed change does not involve a significant hazards consideration, a significant change in the types of, or significant increase in the amounts of any effluents that may be released offsite, or a significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10CFR51.22(c) (9). Therefore, pursuant to 10CFR51.22(b), an environmental assessment of the proposed change is not required.