ENCLOSURE 2 to TXX-07106

WCAP-16840-NP
Comanche Peak Nuclear Power Plant
Stretch Power Uprate Licensing Report
(Non-proprietary)

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Revision 0

August 2007

Comanche Peak Nuclear Power Plant Stretch Power Uprate Licensing Report

Westinghouse

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Comanche Peak Nuclear Power Plant Stretch Power Uprate Licensing Report

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August 2007

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List of Acronyms

ABN	abnormal operating procedures
AC	alternating current
ADV	atmospheric dump valve
AFW	auxiliary feedwater
AFWS	auxiliary feedwater system
AISC	American Institute of Steel Construction
AOO	anticipated operational occurrence
AOR	Analysis of Record
AOV	air-operated valves
AMSAC	ATWS mitigation system actuation circuitry
ANS	American Nuclear Society
ANSI	American National Standards Institute
ART	adjusted reference temperature
ARV	atmospheric relief valve
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ASTRUM	automated statistical treatment of uncertainty method
ATWS	anticipated transient without scram
AVB	anti-vibration bar
BACC	boric acid corrosion control
B&PV	Boiler and Pressure Vessel
BELOCA	best-estimate LOCA
BOL	beginning of life
BOP	balance of plant
BEBF	best-estimate bypass flow
BIT	boron injection tank
BL	bulletin
BMI	bottom-mounted instrumentation
BRS	boron recycle system
BTP	Branch Technical Position
BTRS	boron thermal regeneration system
CASS	cast austenitic stainless steel
CC	component cooling
CCP	centrifugal charging pump
CCW	component cooling water
CDF	core damage frequency
CFD	computational fluid dynamic
CHF	critical heat flux
CIPS	crud-induced power shift
CLB	current licensing basis
CCS	component cooling water system
CF	chemistry factor
CLOF	complete loss of flow
CLOF-FD	complete loss of flow-frequency decay

List of Acronyms (cont.)

CLOF-UV	complete loss of flow-undervoltage
COLR	Core Operating Limit Report
COMS	cold overpressure mitigation system
CPNPP	Comanche Peak Nuclear Power Plant
CPT	critical power trajectory
CQD	Code Qualification Document
CRDM	control rod drive mechanism
CRDS	control rod drive system
CS	chemical and volume control system
CSAU	code scaling, applicability, and uncertainty
CST	condensate storage tank
CT	containment spray system
CUF	cumulative usage factor
CVCS	chemical and volume control system
CW	circulating water
CWO	core-wide oxidation
DBA	design basis accident
DBD	design basis document
DBLOCA	design basis LOCA
DC	direct current
DEHL	double-ended hot leg
DEPS	double-ended pump suction
DER	double-ended rupture
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
DOT	Department of Transportation
DRLL	dropped rod limit line
DSS	diverse scram system
EAB	exclusion area boundary
ECC	emergency core cooling
ECCS	emergency core cooling system
EDY	effective degradation year
EFPD	effective full-power day
EFPY	effective full-power year
EHC	electro-hydraulic control
EM	evaluation model
EOC	end of cycle
EOL	end of life
EOLR	end of license renewal
EOP	emergency operating procedure
EPRI	Electric Power Research Institute
EQ	environmental qualification
EQ	equipment qualification.

List of Acronyms (cont.)

ERCOT	Electric Reliability Council of Texas
ESF	engineered safety features
ESFAS	engineered safety features actuation system
F&O	facts & observations
FAC	flow accelerated corrosion
FCV	flow control valve
FF	fluence factor
FHAR	Fire Hazards Analysis Report
FIV	flow-induced vibration
FIV	feedwater isolation valve
FMEA	failure modes and effects analysis
FPP	fire protection program
FPR	Fire Protection Report
FSAR	Final Safety Analysis Report
FSSAR	Fire Safe Shutdown Analysis Report
FWLW	feedwater line break
GDC	general design criterion
GL	generic letter
GWS	gaseous waste system
HEI	Heat Exchange Institute
HELB	high energy line break
HEPA	high efficiency particulate absorption
HFP	hot full power
HFF	hydraulic forcing function
HHSI	high head safety injection
HI	heat injection
HLSO	hot leg switchover
HP	high pressure
HPO	hot pump overspeed
HRA	human reliability analysis
HVAC	heating, ventilation, and air conditioning
HX	heat exchanger
HZP	hot zero power
I&C	instrumentation and control
IASCC	irradiation-assisted stress corrosion cracking
ID	inner diameter
IEEE	Institute of Electrical and Electronic Engineers
IFBA	integral fuel burnable absorber
IGSCC	inter-granular stress corrosion cracking
IFM	intermediate flow mixer
IGSCC	intergranular stress corrosion cracking
IPE	individual plant examination

List of Acronyms (cont.)

IPEEE	individual plant examination of external events
ISI	in-service inspection
ISLOCA	interfacing system LOCA
IST	in-service testing
LAR	License Amendment Request
LBB	leak before break
LBBELOCA	large-break best-estimate LOCA
LBLOCA	large-break LOCA
LERF	large early release frequency
LHFF	LOCA hydraulic forcing functions
LHSI	low head safety injection
LOCA	loss-of-coolant accident
LOL	loss of load
LONF	loss of normal feedwater
LOOP	loss-of-offsite power
LPZ	low population zone
LR	Licensing Report
LSSS	limiting safety system setting
LTOP	low-temperature overpressure protection
LWR	light water reactor
M&E	mass and energy
M&TE	measurement and test equipment
MCO	moisture carryover
MDAFW	motor-driven auxiliary feedwater
MMF	minimum measured flow
MSSV	main steam safety valve
MDC	moderator density coefficient
MI	mass injection
MMF	minimum measured flow
MRP	Material Reliability Project
MSIV	main steam isolation valve
MSLB	main steam line break
MSR	moisture separator reheater
MSS	main steam system
MSSV	main steam safety valve
MTC	moderator temperature coefficient
MUR	measurement uncertainty recapture
NDE	non-destructive examination
NEI	Nuclear Energy Institute
NEMA	National Electrical Manufacturer's Association
NFPA	National Fire Protection Association
NOP	normal operating pressure

List of Acronyms (cont.)

NPSH	net positive suction head
NRC	Nuclear Regulatory Commission
NRS	narrow-range span
NRVs	non-return check valves
NSSS	nuclear steam supply system
NUPPSCO	Nuclear Power Plant Standards Committee
OBE	operating basis earthquake
ODCM	Offsite Dose Calculation Manual
OFA	optimized fuel assembly
P&ID	piping and instrumentation diagram
PCT	peak cladding temperature
PCWG	Performance Capabilities Working Group
PLOF	partial loss of flow
PMP	probable maximum precipitation
PORV	power-operated relief valve
PTLR	Pressure and Temperature Limits Report
PRA	probabilistic risk assessment
PRT	pressurizer relief tank
PSV	pressurizer safety valve
P-T	pressure-temperature
PTS	pressurized thermal shock
PWR	pressurized water reactor
PWROG	Pressurized Water Reactor Owners Group
PWSCC	primary water stress corrosion cracking
QA	quality assurance
RAI	request for additional information
RCCA	rod cluster control assembly
RCL	reactor coolant loop
RCP	reactor coolant pump
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RG	Regulatory Guide
RHR	residual heat removal
RHRS	residual heat removal system
RMS	root mean square
RMS	radiation monitoring systems
RMWS	reactor makeup water system
RPS	reactor protection system
RPV	reactor pressure vessel
RRVCH	replacement reactor vessel closure head
RSB 5-1	Reactor System Branch Technical Position 5-1
RSE	reload safety evaluation

List of Acronyms (cont.)

RSG	replacement steam generator
RTD	resistance temperature detector
RTDP	Revised Thermal Design Procedure
RTP	rated thermal power
RTS	reactor trip system
RV	reactor vessel
RVCH	reactor vessel closure head
RVHP	reactor vessel head penetrations
RVI	reactor vessel internal
RVLIS	reactor vessel level instrumentation system
RWAP	rod withdrawn at power
RWFS	rod withdrawal from subcritical
RWST	refueling water storage tank
SAFDL	specified acceptable fuel design limit
SAL	safety analysis limit
SBLOCA	small-break LOCA
SBO	station blackout
SCC	stress corrosion cracking
SCR	Squaw Creek Reservoir
SEP	Systematic Evaluation Program
SER	Safety Evaluation Report
SF	spent fuel
SFP	spent fuel pool
SG	steam generator
SGB	steam generator blowdown
SGDD	Steam Generator Degradation Database
SGTP	steam generator tube plugging
SGTR	steam generator tube rupture
SGWLC	steam generator water level control
SI	safety injection
SIS	safety injection signal
SFP	spent fuel pool
SPU	stretch power uprate
SRP	Standard Review Plan
SRSS	square root sum of the squares
SSC	structure, system, and component
SSE	safe shutdown earthquake
SSER	Supplemental Safety Evaluation Report
SSI	safe shutdown impoundment
STDP	Standard Thermal Design Procedure
T-H	thermal-hydraulic
TC	thermocouple
TDAFW	turbine-driven auxiliary feedwater

List of Acronyms (cont.)

TDBF	thermal design bypass flow
TDF	thermal design flow
TEDE	total effective dose equivalent
TGSCC	transgranular stress corrosion cracking
TPI	thimble plugs installed
TSSCC	transgranular stress corrosion cracking
TT	turbine trip
T/H	thermal-hydraulic
THRIVE	Thermal Hydraulic Reactor Internals Vessel Evaluation
TLAA	time-limited aging analyses
TMI	Three Mile Island
TPI	thimble plug installed
TPS	tube support plate
TS	Technical Specification
TW	through-wall
UET	unfavorable exposure time
UHS	ultimate heat sink
UPI	upper plenum injection
UPS	uninterruptible power supply
USE	upper shelf energy
VAR	volt-ampere reactive
VCT	volume control tank
WOG	Westinghouse Owners Group
1-D	one-dimensional
2-D	two-dimensional
3-D	three-dimensional

1.0 INTRODUCTION

General Overview

This Licensing Report (LR) is provided by TXU Generation Company LP (herein after referred to as Luminant Power) in support of the Unit 1 and Unit 2 Comanche Peak Steam Electric Station (herein after referred to as Comanche Peak Nuclear Power Plant (CPNPP)) stretch power uprate (SPU) license application.

The LR is a summary of the analyses and evaluations performed to demonstrate that the proposed increase in power can be safely achieved with no adverse impact on the health and safety of the public. The LR is an attachment to the license amendment request (LAR), which requests Nuclear Regulatory Commission (NRC) approval to increase the licensed core power level of both CPNPP units from 3,458 to 3,612 MWt. The LR provides the details that support the requested license and technical specification changes and works in concert with the other attachments to the amendment request to provide reviewers with a comprehensive evaluation of the effects of the proposed SPU.

Both CPNPP units have previously implemented a 1.4-percent measurement uncertainty recapture (MUR) uprate. Based on NRC SECY 2001-0124, SPUs typically involve power increases up to 7 percent above the plant's original licensed power level and they do not generally involve major plant modifications. SECY 2001-0124 notes that in some limited cases where plant equipment was operated near capacity prior to the SPU, more substantial changes may be required. For both CPNPP Units, the proposed core power level of 3,612 MWt will result in operation at 5.9 percent above the original licensed core power of 3,411 MWt and approximately 4.5 percent above the current licensed core power level of 3,458 MWt. The plant modifications planned in conjunction with this uprate application are listed in Table 1.0-1 and are further described later in this section. Table 1.0-1 provides a summary of the modifications that are needed to support the uprate, and those that are being made as a plant enhancement. The modifications are not considered to be major modifications as most are component level upgrades where the overall system design and configuration remains unchanged. The safety related modifications are considered to be minor modifications to pipe supports, setpoints and control settings only. Two non-safety related modifications (high pressure turbine upgrade and main transformer replacement) would be considered major from a capital cost standpoint, but from a plant design and implementation standpoint they are similar to modifications that have previously been successfully implemented at CPNPP. Based on the above discussion and the supporting information in this submittal, Luminant Power considers this uprate application as a SPU application.

Nonetheless, the CPNPP uprate evaluations in this LR address the scope and follow the format of the NRC's extended power uprate review standard, RS-001 "Review Standard for Extended Power Uprates."

Technical evaluations presented in the LR include, when appropriate, discussion of the effects of SPU on plant operating limits, functional performance requirements and design margins. The

LR also describes the methodology utilized where different than those used in the current licensing basis.

The technical evaluations contained in the LR address both CPNPP Units 1 and 2. For the most part, a single analysis that bounds both units is presented since the two units are almost identical. In cases where substantive differences exist, separate evaluations are performed and described in the appropriate LR section.

Many plants similar to CPNPP have already implemented SPUs. One example is Seabrook Station which has recently implemented a similar uprate to a core thermal power of 3,648 MWt, 36 MWt greater than the proposed SPU for CPNPP. Seabrook Station and CPNPP (Unit 1 and Unit 2) are four-loop Westinghouse nuclear steam supply system (NSSS) plants. A comparison of the significant NSSS design parameters for both the uprated CPNPP units and the current Seabrook Station is provided in Table 1.0-2.

The LR development process included consideration of recent power uprate NRC Requests for Additional Information (RAI).

Schedule

Modifications necessary to allow operation at the SPU conditions will be implemented before or during the Fall 2008 outage for Unit 1 and the Fall 2009 outage for Unit 2. Subject to NRC approval of this license application, power ascension to 3,612 MWt is planned for startup from the Fall 2008 refueling outage for Unit 1 (1RF13) and startup from the Fall 2009 refueling outage for Unit 2 (2RF11).

Plant Modifications

Table 1.0-1 provides a listing of the plant physical modifications required to support operation at the uprate power level. The following discussion highlights the principal design and modification changes associated with the SPU.

<u>Fuel/Reactor Core Design</u>

The uprated core will operate at a core thermal power of 3,612 MWt as compared to the current core thermal power of 3,458 MWt. This represents an increase of approximately 4.5 percent in core thermal power. No change in the fuel assembly design is required nor proposed for SPU. The core operating limits will continue to be established using NRC-approved methodologies, and all fuel design constraints will continue to be satisfied. Several setpoints changes for the reactor trip system and the engineering safety features actuation system will be revised as part of License Amendment Request (LAR) 07-006 (TXX-07108). These setpoints remain valid for operation at the SPU conditions, and no additional changes are required to ensure the fuel design limits are maintained.

Reactor Coolant System

There are no physical modifications planned to the reactor coolant system or reactor vessel internals.

The reactor coolant system operation will change very little for the uprate. The coolant temperature increase across the core will approximately increase in proportion to the increase in uprate power. With the higher core temperature difference, the reactor vessel outlet temperature (T_{hot}) will increase by slightly more than 1°F. The reactor coolant system no-load temperature will remain at the current value of 557°F.

Steam Generator and Main Steam

The Unit 1 steam generators were replaced in 2007 with Westinghouse Δ76 steam generators. The Unit 2 steam generators are original D-5 steam generators.

For Unit 1, the best estimate steam generator steam pressure for the uprate will remain at 1,000 psia due to a slight increase in T_{avg}. For Unit 2, the same T_{avg} will be used at the uprated conditions, which results in a slightly lower best estimate steam generator steam pressure of approximately 978 psia.

Main Turbine

The Units 1 and 2 high pressure turbines will be replaced in order to pass the additional volumetric steam flow. Turbine digital controls and thyristor voltage regulator settings will be revised for uprate conditions. The low pressure turbines will not be modified as they are capable of passing the higher volumetric flow rate.

Main Condenser/Circulating Water/Squaw Creek Reservoir

The Squaw Creek Reservoir temperature will increase about 0.7°F due to uprate of both Units 1 and 2. The circulating water temperature rise through the condenser will also increase approximately 0.7°F. As a result, the temperature of the circulating water discharged to Squaw Creek Reservoir will increase about 1.5°F. No change is required to the Texas Pollution Discharge Elimination System (TPDES) permit.

Condensate and Feedwater

The condensate and feedwater flow rates will increase approximately in proportion to the uprate power increase. The condensate pumps and main feedwater pumps (MFPs) are not required to be modified.

Higher condensate pump flow rate and additional head loss in the condensate and feedwater piping will result in lower suction pressure at the MFP. To preserve operating margin to alarms and automatic actions on low MFP suction pressure, the setpoints associated with MFP net

positive suction head (NPSH) protection, condensate polisher bypass, and feedwater heater bypass will be changed.

Extraction Steam and Heater Drains

There will be slight increases in the temperatures, pressures and flows in the extraction steam piping and in the various heater drains. Modifications to increase the capacity of the heater drain pumps will be installed to satisfy uprate heater drain flow requirements.

Main Generator

The main generator electrical output will increase by approximately 49 MWe (Unit 1) and 37 MWe (Unit 2). Each main generator will be re-rated from 1,350 to 1,410 MVA with an allowable power factor of 0.9.

The hydrogen coolers will be replaced and additional cooling will be provided to the exciter air coolers for additional cooling in the summer months.

Iso-Phase Bus Ducts/Main Transformers

To transfer the power from the main generator to the grid the design capacity of the iso-phase bus duct system will be increased. The bus duct cooling fan/coil capacity will be increased to provide additional cooling.

The main transformers are currently operating under administrative voltage limits. The main transformers have been evaluated and found acceptable at SPU conditions with the current administrative limits. The main transformers are scheduled to be replaced due to their age and to enhance their MVAR support capability. Unit 2 transformers are scheduled to be replaced in 2009 and Unit 1 in 2010, after one cycle of SPU operation.

Spent Fuel Pool Storage

The current licensing basis allows for storage of up to 3,373 spent fuel assemblies in two storage pools in a common fuel building. The fuel assembly storage locations are comprised of two different designs (Region I and Region II storage racks). Each pool may be used to store fuel from either or both of the CPNPP units.

Spent Fuel Pool 1 and Spent Fuel Pool 2 (SFP1 and SFP2) contain 222 and 219 Region I storage racks, respectively (441 total). The Region I racks are designed to accommodate new fuel with a maximum enrichment of 5.0 w/t percent U-235 or spent fuel regardless of the discharge fuel burnup. Soluble boron is not credited for the storage of spent fuel assemblies within the Region I racks, and there are no storage pattern restrictions associated with the Region I racks. The neutron absorber material Boral is credited for the storage of spent fuel assemblies within the Region I racks to maintain k_{eff} less than or equal to 0.95.

The Region II rack design in SFP1 and SFP2 permit four different configurations of storage, which, for the purpose of criticality considerations, are considered as separate pools. Region II racks, with 1,462 and 1,470 storage positions in SFP1 and SFP2 respectively (2,932 total), are designed to accommodate fuel of various initial enrichments which have accumulated minimum burnups and decay times within either (1) the "acceptable" domain of a 4 out of 4 configuration, (2) the "acceptable" domain of a 3 out of 4 configuration, (3) the "acceptable" domain of a 2 out of 4 configuration, or (4) a 1 out of 4 configuration. Operation at SPU conditions affects the reactivity of discharged fuel, therefore Technical Specification 3.7.17 changes are included in the SPU submittal. See LR Section 2.8.6.2.

At the current licensed power level, there is sufficient storage capacity through 2013. The SPU analyses and the associated Technical Specification changes discussed above result in an increased capability of the SFP due to reactivity credits of plutonium decay and axial blankets.

Additional Review Areas

In order to provide a complete description of the analysis performed, the licensing report takes advantage of the provision in RS-001 to add additional sections (additional review areas). The following sections are in addition to the standard template:

1.0	Introduction
1.1	Nuclear Steam Supply System Parameters
2.2.6	NSSS Design Transients
2.2.7	Bottom-Mounted Instrumentation and Guide Tubes and Flux Thimbles
2.4.2	Plant Operability/Component Sizing
2.5.8.1	Circulating Water System
2.7.7	Other Ventilation Systems (Containment)
2.8.5.0	Non-LOCA Analysis Introduction
2.8.7.1	Auxiliary Systems Pumps, Heat Exchangers, Valves, and Tanks
2.8.7.2	Natural Circulation Cooldown
2.8.7.3	Loss of Residual Heat Removal at Mid-Loop
2.9.10	Steam Releases from Intact Steam Generators for Locked Rotor and MSLB Radiological Dose Analyses
2.9.11	Radiological Consequences of Gas Decay Tank Rupture
2.9.12	Radiological Consequences of Liquid Waste Tank Rupture
Appendix A	Codes and Methods

SPU Relation to Other Recent and Concurrent Licensing Activities

The following licensing activities were being evaluated during development of the SPU submittal. The below discussion summarizes how the SPU is coordinated with these other activities.

Transition to Westinghouse Methodology

License Amendment Request LAR-07-003 was submitted in April 2007. The safety analyses performed for uprate have been performed with the Westinghouse accident analysis methodologies. As such, approval of the Westinghouse accident analysis methodologies for application at CPNPP is required to support the SPU. Cycle specific Technical Specification changes are necessary as a result of analyses performed with new methods. The analyses were performed at uprate conditions and resulting licensing changes have been submitted to the NRC as cycle specific license amendment requests (LARs) for Unit 1 and Unit 2. In such instances, the affected license report section will refer to the cycle specific LARs.

Turbine Trip Valve Test Interval

License Amendment Request LAR-07-001 was submitted in May 2007. The uprate turbine missile analysis assumes the 26 week test interval proposed in LAR-07-001. As such, approval of the proposed 26 week test interval is assumed in this uprate submittal.

Treatment of Proprietary Information Referenced Within the Licensing Report

Two versions of the LR have been prepared: proprietary and non-proprietary. The non-proprietary version is for placement within the public document room. The proprietary version is for use by the NRC reviewers. The affidavit describing the nature of the proprietary information is provided in the SPU license amendment request. Every effort was made to minimize the amount of information withheld. Bracketed "[ ]" information designates data that is Westinghouse Proprietary.

Table 1.0-1

CPNPP Unit 1 and Unit 2 SPU Planned Modifications

#	Modification	Category (A, B, C)[1]	LR Section
1	NSSS setpoints, settings, scaling	A	2.4.1
2	Pipe support modifications	A, B	2.2.2.2
3	HP Turbine Upgrade	B	2.5.1.2.2
4	Turbine Digital Controls and Voltage Regulator Setpoints	B	2.5.1.2.2
5	Heater Drain Pump Rotating Element	B	2.5.5.4
6	Main Generator Hydrogen Coolers Replacement	B	2.3.3
7	Iso-Phase Bus Duct Coolers Replacement	B	2.3.3
8	Turbine Plant Cooling Water (TPCW) modifications for Main Generator Exciter Air Coolers	B	2.3.3
9	BOP Setpoints, Settings, Scaling	B	2.3.3 / 2.4.1
10	Main Step Up Transformers Replacement	C	2.3.3

Notes:

1. Category Key
   - A. Required to support safety analyses
   - B. Required to support operation at uprated power level
   - C. Enhancement, not essential for uprate operation

Table 1.0-2

Comparison of CPNPP and Seabrook NSSS Design Parameters

Parameter	CPNPP Unit 1 and 2		Seabrook Uprate Values
	Current Values	SPU Values	
Reactor Core Power (MWt)	3,458	3,612	3,648
Fuel Type	VANTAGE+	VANTAGE+	RFA with IFMs
Vessel Average Coolant Temp. Hot Full Power (HFP) (°F)	U1: 574.2 to 589.2 U2: 589.2	574.2 to 589.2	571.0 to 589.1
Maximum Core Inlet Temperature (°F)	U1: 559.2 U2: 562.7	558.0	556.8
Maximum Core Outlet Temperature (°F)	U1: 622.5 U2: 625.0	623.8	626.5
Coolant System Pressure (psia)	2,250	2,250	2,250
Thermal Design Flow (gpm/loop)	U1: 95,700 U2: 97,900	95,700	93,600

1.1 NUCLEAR STEAM SUPPLY SYSTEM PARAMETERS

1.1.1 Introduction

The nuclear steam supply system (NSSS) design parameters are the fundamental parameters used as input in all of the NSSS analyses. The current Comanche Peak Nuclear Power Plant (CPNPP) Units 1 and 2 NSSS design parameters are summarized in Table 5.1-1 of the CPNPP Units 1 and 2 Final Safety Analysis Report (FSAR). The NSSS design parameters provide the primary and secondary side system conditions (thermal power, temperatures, pressures, and flows) that serve as the basis for all of the NSSS analyses and evaluations. As a result of the stretch power uprate (SPU) program, the CPNPP Units 1 and 2 NSSS design parameters have been revised, as shown in Tables 1.1-1 and 1.1-2. Tables 1.1-1 and 1.1-2 provide information for the four cases associated with the SPU. These parameters have been incorporated, as appropriate, into the applicable NSSS systems and components evaluations, as well as safety analyses, performed in support of the SPU.

1.1.2 Input Parameters, Assumptions, and Acceptance Criteria

The NSSS design parameters, also referred to as the Performance Capability Working Group (PCWG) parameters, provide the reactor coolant system (RCS) and secondary system conditions (thermal power, temperatures, pressures, and flows) that are used as the basis for the NSSS design transients, systems, structures, components, accidents, and fuel analyses and evaluations.

The computer code used to determine the NSSS design parameters is NSSSPlus (formally SGPER). The code and method used to calculate these values have been successfully used to license all previous similar programs for Westinghouse plants and employ basic thermal-hydraulic calculations, along with first principles of engineering, to generate the temperatures, pressures, and flows shown in Tables 1.1-1 and 1.1-2.

The major input parameters and assumptions used in the calculation of the four cases of NSSS design parameters established for the uprate program are summarized as follows:

- The parameters are based on Westinghouse Model Δ76 steam generators for CPNPP Unit 1 and Westinghouse Model D-5 steam generators for CPNPP Unit 2.
- The uprated thermal power of 3,612 MWt is approximately 4.5-percent higher than the current licensed power of 3,458 MWt and approximately 5.9-percent higher than the original licensed power of 3,411 MWt.
- The uprated NSSS power level of 3,628 MWt (3,612 MWt core power + 16 MWt net heat input) was assumed.
- A nominal feedwater temperature (T_{feed}) range of 390.0° to 450.3°F was selected.

- The design core bypass flow was assumed to be 5.8 percent; this accounts for fuel with thimble plugs installed (TPI) and intermediate flow mixing vane (IFM) grids.

- A thermal design flow (TDF) of 95,700 gpm/loop was used.

- A full-power normal operating vessel average temperature (T_{avg}) range of 574.2° to 589.2°F was assumed. This provides the basis for CPNPP Units 1 and 2 to operate within this window. Any exceptions to these values will be addressed in the affected sections.

- Steam generator tube plugging (SGTP) levels of 0 and 10-percent were assumed.

- A maximum steam generator moisture carryover of 0.10-percent was utilized for CPNPP Unit 1 and 0.25 percent was utilized for CPNPP Unit 2.

Acceptance Criteria

The acceptance criteria for determining the NSSS design parameters were that the results of the SPU analyses and evaluations continue to comply with all CPNPP Units 1 and 2 applicable industry and regulatory requirements, and that they provide CPNPP Units 1 and 2 with adequate flexibility and margin during plant operation.

1.1.3 Description of Analyses and Evaluation

Table 1.1-1 provides the NSSS design parameter cases that were generated and serve as the CPNPP Unit 1 basis for the SPU. These cases are:

- SPU Cases 1 and 2 of Table 1.1-1 represent parameters based on a T_{avg} of 574.2°F. Case 2, which is based on an average 10-percent SGTP, yields the minimum secondary side steam generator pressure and temperature. Note that all primary side temperatures are identical for these two cases.

- SPU Cases 3 and 4 of Table 1.1-1 represent parameters based on the T_{avg} of 589.2°F. Case 3, which is based on 0-percent SGTP, yields the highest secondary side steam generator pressure performance conditions. Note that all primary side temperatures are identical for these two cases. As provided via footnote "(2)" of Table 1.1-1, for instances where an absolute upper limit steam generator outlet pressure is conservative for any analyses, these data are based on the Case 3 parameters with 0-percent SGTP. In addition they assume a steam generator fouling factor of zero.

Best-estimate calculation based performance predictions for CPNPP Unit 1 were also performed for the SPU. These calculations were performed to estimate the actual expected steam conditions at the steam generator outlet as opposed to the design conditions shown in Table 1.1-1. The results were used in the CPNPP Unit 1 balance-of-plant (BOP) analyses performed for the SPU.

Table 1.1-2 provides the NSSS design parameter cases that were generated and serve as the CPNPP Unit 2 basis for the SPU. These cases are:

- SPU Cases 1 and 2 of Table 1.1-2 represent parameters based on a T_{avg} of 574.2°F. Case 2, which is based on an average 10-percent SGTP, yields the minimum secondary side steam generator pressure and temperature. Note that all primary side temperatures are identical for these two cases.

- SPU Cases 3 and 4 of Table 1.1-2 represent parameters based on the T_{avg} of 589.2°F. Case 3, which is based on 0-percent SGTP, yields the highest secondary side steam generator pressure performance conditions. Note that all primary side temperatures are identical for these two cases. As provided via footnote "(2)" of Table 1.1-2, for instances where an absolute upper limit steam generator outlet pressure is conservative for any analyses, these data are based on the Case 3 parameters with 0-percent SGTP. In addition they assume a steam generator fouling factor of zero.

Best-estimate calorimetric measurement based performance predictions for CPNPP Unit 2 were also performed for the SPU. These calorimetric measurement based calculations were performed to estimate the actual expected steam conditions at the steam generator outlet as opposed to the design conditions shown in Table 1.1-2. The calorimetric measurement based calculations used CPNPP Unit 2 plant measured calorimetric data to determine NSSS performance. The results were used in the Unit 2 BOP analyses performed for the SPU.

1.1.4 Conclusions

The resulting PCWG parameters (Tables 1.1-1 and 1.1-2) were used by Westinghouse as the basis for all the NSSS analytical efforts. Westinghouse performed the analyses and evaluations based on the parameter sets that were most limiting, so that the analyses would support operation over the entire range of conditions specified. In cases where the SPU analyses performed do not bound the entire range of conditions specified (such as a restricted T_{avg} operating range), the applicable Licensing Report (LR) section identifies the range of conditions analyzed.

Table 1.1-1
NSSS PCWG Parameters for CPNPP Unit 1 SPU Program

Thermal Design Parameters	Uprate Program Case 1	Case 2	Case 3	Case 4
NSSS Power, MWt	3,628	3,628	3,628	3,628
10^6 Btu/hr	12,379	12,379	12,379	12,379
Reactor Power, MWt	3,612	3,612	3,612	3,612
10^6 Btu/hr	12,325	12,325	12,325	12,325
Thermal Design Flow, loop gpm	95,700	95,700	95,700	95,700
Reactor 10^6 lb/hr	145.5	145.5	142.4	142.4
Reactor Coolant Pressure, psia	2,250	2,250	2,250	2,250
Core Bypass, %	5.8[(1)]	5.8[(1)]	5.8[(1)]	5.8[(1)]
Reactor Coolant Temperature, °F				
Core Outlet	609.8	609.8	623.8	623.8
Vessel Outlet	606.2	606.2	620.4	620.4
Core Average	577.6	577.6	592.8	592.8
Vessel Average	574.2	574.2	589.2	589.2
Vessel/Core Inlet	542.2	542.2	558.0	558.0
Steam Generator Outlet	541.9	541.9	557.6	557.6
Steam Generator				
Steam Outlet Temperature, °F	528.9	526.9	545.1[(2)]	543.1
Steam Outlet Pressure, psia	877	862	1,005[(2)]	988
Steam Outlet Flow, 10^6 lb/hr total	14.89/16.17	14.88/16.16	14.97/16.26[(2)]	14.96/16.25
Feed Temperature, °F	390.0/450.3	390.0/450.3	390.0/450.3	390.0/450.3
Steam Outlet Moisture, % max.	0.10	0.10	0.10	0.10
Tube Plugging Level, %	0	10	0	10
Zero-Load Temperature, °F	557	557	557	557
Hydraulic Design Parameters				
Pump Design Point, Flow (gpm)/Head (ft)	101,000/286			
Mechanical Design Flow, gpm	109,000			
Minimum Measured Flow, gpm total	396,400			

Notes:

1. Core bypass flow accounts for TPI and IFMs.
2. Where appropriate for NSSS analyses, a greater steam pressure of 1,032 psia, steam temperature of 548.4°F, and steam flow of 16.29 x 10^6 lb/hr total is assumed. This envelopes more efficient steam generator performance.

Table 1.1-2

NSSS PCWG Parameters for CPNPP Unit 2 SPU Program

Thermal Design Parameters	Uprate Program			
	Case 1	**Case 2**	**Case 3**	**Case 4**
NSSS Power, MWt	3,628	3,628	3,628	3,628
10^6 Btu/hr	12,379	12,379	12,379	12,379
Reactor Power, MWt	3,612	3,612	3,612	3,612
10^6 Btu/hr	12,325	12,325	12,325	12,325
Thermal Design Flow, loop gpm	95,700	95,700	95,700	95,700
Reactor 10^6 lb/hr	145.5	145.5	142.4	142.4
Reactor Coolant Pressure, psia	2,250	2,250	2,250	2,250
Core Bypass, %	5.8[1]	5.8[1]	5.8[1]	5.8[1]
Reactor Coolant Temperature, °F				
Core Outlet	609.8	609.8	623.8	623.8
Vessel Outlet	606.2	606.2	620.4	620.4
Core Average	577.6	577.6	592.8	592.8
Vessel Average	574.2	574.2	589.2	589.2
Vessel/Core Inlet	542.2	542.2	558.0	558.0
Steam Generator Outlet	541.9	541.9	557.6	557.6
Steam Generator				
Steam Outlet Temperature, °F	525.8/522.6	522.0/518.7	543.2/539.7[2]	539.4/535.9
Steam Outlet Pressure, psia	854/831	826/804	989/961[2]	958/930
Steam Outlet Flow, 10^6 lb/hr total	14.90/16.17	14.89/16.15	14.99/16.26[2]	14.97/16.24
Feed Temperature, °F	390.0/450.3	390.0/450.3	390.0/450.3	390.0/450.3
Steam Outlet Moisture, % max.	0.25	0.25	0.25	0.25
Tube Plugging Level, %	0	10	0	10
Zero Load Temperature, °F	557	557	557	557
Hydraulic Design Parameters				
Pump Design Point, Flow (gpm)/Head (ft)	101,000/286			
Mechanical Design Flow, gpm	105,000			
Minimum Measured Flow, gpm total	408,000			

Notes:

1. Core bypass flow accounts for TPI and IFMs.
2. Where appropriate for NSSS analyses, a greater steam pressure of 1,017 psia, steam temperature of 546.6°F, and steam flow of 15.01 x 10^6 lb/hr total should be assumed. This envelopes more efficient steam generator performance.

2.1 MATERIALS AND CHEMICAL ENGINEERING

2.1.1 Reactor Vessel Material Surveillance Program

2.1.1.1 Regulatory Evaluation

The reactor vessel (RV) material surveillance program provides a means for determining and monitoring the fracture toughness of the RV beltline materials to support analyses for ensuring the structural integrity of the ferritic components of the RV. Luminant Power's review focused on the effects of the uprate on the present RV surveillance capsule withdrawal schedule.

The acceptance criteria are based on:

- General Design Criterion (GDC)-14, insofar as it requires that the reactor coolant pressure boundary (RCPB) be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture.
- GDC-31, insofar as it requires that the RCPB be designed with margin sufficient to ensure that, under specific conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized.
- 10 CFR Part 50, Appendix H, which provides for monitoring changes in the fracture toughness properties of materials in the RV beltline region.
- 10 CFR Part 50.60, which requires compliance with the requirements of 10 CFR Part 50, Appendix H.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the CPNPP reactor protection, engineered safety feature actuation, and control systems for adequacy regarding conformance to:

- GDC-14, Reactor Coolant Pressure Boundary, is described in FSAR Section 3.1.2.5.

  The reactor coolant system (RCS) pressure boundary is designed to accommodate the system pressures and temperatures attained under the expected modes of plant operation, including anticipated transients, and to maintain the stresses within applicable stress limits (FSAR Section 5.2). Also, RCPB material and selection and fabrication techniques ensure a low probability of gross rupture of significant leakage.

In addition to the loads imposed on the system under normal operating conditions, consideration is also given to abnormal loading conditions, such as seismic and pipe rupture, which are discussed in FSAR Sections 3.6 and 3.7.

The RCS boundary has provisions for inspection, testing, and surveillance of critical areas to assess the structural and leak-tight integrity (FSAR Section 5.2). For the RV, a material surveillance program conforming to applicable codes is provided (FSAR Section 5.3).

- GDC-31, Fracture Prevention of Reactor Coolant Pressure boundary, is described in FSAR Section 3.1.4.2.

  Close control is maintained over material selection and fabrication for the RCS to ensure that the boundary behaves in a nonbrittle manner. Those RCS materials that are exposed to the coolant are corrosion resistant stainless steel or Inconel. The reference temperature (RT_{NDT}) of the RV structural steel is established by the Charpy V-notch and drop weight tests in accordance with 10 CFR Part 50, Appendix G.

2.1.1.2 Technical Evaluation

2.1.1.2.1 Introduction

Reactor vessel integrity is impacted by any change in plant parameters that affect neutron fluence levels or temperature/pressure transients. The changes in neutron fluence resulting from the stretch power uprate (SPU) have been evaluated to determine the impact on RV integrity. The assessment presented herein focuses on the surveillance capsule withdrawal schedules for CPNPP Units 1 and 2 contained in the most recent surveillance capsule evaluations – WCAP-16610 (Reference 1) for Unit 1 and WCAP-16277 (Reference 2) for Unit 2. In this section, vessel fluence values are used to evaluate the transition temperature shift (ΔRT_{NDT}) to confirm the validity of the surveillance capsule withdrawal schedules previously established.

2.1.1.2.2 Input Parameters, Assumptions, and Acceptance Criteria

SPU Fluence Projections

The fast neutron (E > 1.0 MeV) fluence experienced by the materials comprising the beltline region of the reactor pressure vessel is used as input in all pressure vessel integrity evaluations that involve an assessment of radiation-induced degradation in material properties. In determining the fast neutron fluence for the reactor materials, a plant-specific calculation is normally performed for fuel cycles that have been completed and fluence projections for future operation are generated based on an assumed mode of operation. The key parameters in choosing a future mode of operation are the assumed spatial distribution of the neutron source within the reactor core and the core power level. Analyses of this type were completed coincident with the last ex-vessel neutron dosimetry withdrawal from CPNPP Unit 1 and the last in-vessel surveillance capsule withdrawal from CPNPP Unit 2.

All of the neutron exposure evaluations for CPNPP Units 1 and 2 used an NRC approved methodology that meets the requirements of Regulatory Guide 1.190 (Reference 3). The approval is documented in WCAP-14040 (Reference 4). The neutron fluence calculations for the pressure vessel materials were based on the synthesis methodology described in Reference 4 and were completed using the DORT two-dimensional discrete ordinates code from the DOORS 3.2 Code Package (Reference 5). The transport cross-sections used in the calculations were taken from the BUGLE-96 coupled neutron/photon cross-section library (Reference 6) that was generated specifically for light water reactor (LWR) applications and has been evaluated as an integral part of the approved fluence methodology.

Results of the fast neutron fluence evaluation for CPNPP Units 1 and 2 are provided in Tables 2.1.1-1 through 2.1.1-3.

Inlet Temperature

As presented in Licensing Report (LR) Section 1.1, Nuclear Steam Supply System Parameters, the SPU full-power RV inlet temperature range is 542.2° to 558.0°F.

Chemistry Factor Values

The chemistry factors (CFs), along with the fluence factors (FFs), are used to determine ΔRT_{NDT}. The CFs used in this evaluation are presented in Tables 2.1.1-4, along with the best-estimate copper and nickel chemistry used to calculate the CF values.

Transition Temperature Shift Values

Results of the ΔRT_{NDT} calculations for each of the plates and welds in the CPNPP Units 1 and 2 RV beltline are presented in Table 2.1.1-5.

Acceptance Criteria

The acceptance criteria for performing material surveillance of the RV and for generating a withdrawal schedule are in 10 CFR Part 50, Appendix H and American Society for Testing and Materials (ASTM) E 185-82, Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels, E 706.

The acceptance criteria for the RV inlet temperature are provided in U.S. NRC Regulatory Guide 1.99, Revision 2, Radiation Embrittlement of Reactor Vessel Materials. This states that "The procedures are valid for a nominal irradiation temperature of 550°F. Irradiation below 525°F should be considered to produce greater embrittlement, and irradiation above 590°F may be considered to produce less embrittlement." Therefore, the RV inlet temperature must be greater than 525°F and less than 590°F for the equations and methodology of Regulatory Guide 1.99, Revision 2 to remain valid.

2.1.1.2.3 Description of Analyses and Evaluations

The RV surveillance capsule withdrawal schedule evaluation for the proposed CPNPP Units 1 and 2 SPU includes a review of the RV inlet temperature to verify that it complies with Regulatory Guide 1.99, Revision 2, and a review of the vessel fluence projections to determine if changes are required to the number of capsules withdrawn and/or the schedule for withdrawal. This evaluation is consistent with the recommended practices of ASTM E 185-82 and meets the requirements of 10 CFR 50, Appendix H.

A surveillance capsule withdrawal schedule was developed to periodically remove surveillance capsules from the RV in order to effectively monitor the condition of the RV materials under actual operating conditions. ASTM E 185-82 defines the recommended number of surveillance capsules, and the recommended withdrawal schedule, based on the predicted transition temperature shifts (ΔRT_{NDT}) of the vessel material. The surveillance capsule withdrawal schedule is in terms of effective full-power years (EFPY) of plant operation with an original design life of 32 EFPYs. Other factors considered in establishing the surveillance capsule withdrawal schedule are the maximum fluence values at the vessel surface and the approval of life extension to a new design life.

The first surveillance capsule is usually scheduled to be withdrawn early in the vessel life to verify the initial predictions of the surveillance material response to the actual radiation environment. It is generally removed when the predicted shift exceeds the expected scatter by a sufficient margin to be measurable. Normally, the capsule with the highest lead factor is withdrawn first. Early withdrawal also permits verification of the adequacy and conservatism of the RV pressure-temperature (P-T) operation limits. The withdrawal schedule for the remaining surveillance capsules to be withdrawn was adjusted by the lead factor so that:

- The neutron fluence exposure of the second surveillance capsule corresponds to the original design life 32 EFPY fluence at the RV inner wall location.
- The exposure of the third surveillance capsule withdrawn exceeds the peak design life vessel fluence (now estimated to be 36 EFPYs), but does not exceed twice that value.

Per ASTM E 185-82, the four steps used for the development of a surveillance capsule withdrawal schedule are as follows:

- Estimate the peak vessel inside surface fluence at end of life and the corresponding transition temperature shift. This identifies the number of capsules required. Per Regulatory Guide 1.99, Revision 2, the transition temperature shift (ΔRT_{NDT}) is equal to the CF times the FF. In the case of determining the number of capsules to be withdrawn, the peak vessel surface fluence is used to determine the FF.
- Obtain the lead factor for each surveillance capsule relative to the peak beltline fluence.

- Calculate the EFPY for the capsule to reach the peak vessel end-of-life fluence at the inside surface. These are used to establish the withdrawal schedule for all but the first surveillance capsule.

- Schedule the surveillance capsule withdrawals at the nearest vessel refueling date.

A surveillance capsule withdrawal schedule was developed for the CPNPP RVs and documented in WCAP-16610 (Reference 1) for Unit 1 and WCAP-16277 (Reference 2) for Unit 2. Updated neutron fluence projections are utilized herein to evaluate the applicability of those withdrawal schedules for CPNPP Units 1 and 2 in the licensing bases for the SPUs.

2.1.1.2.4 Results

Reactor vessel fluence projections were generated for the SPU conditions following the guidance of Regulatory Guide 1.190 (presented in Table 2.1.1-1). Note that these SPU vessel fluence projections are lower than the vessel fluence projections documented in References 1 and 2. Chemistry factors for each of the beltline materials were determined in accordance with Regulatory Guide 1.99, Revision 2, Positions 1.1 and 1.2, as presented in Table 2.1.1-4. Transition temperature shifts were then calculated for each of the beltline materials (vessel inside surface) to determine the appropriate surveillance capsule withdrawal schedule (Table 2.1.1-5). The calculations were performed at 36 EFPY, and the maximum neutron exposure for the beltline materials at each unit were applied to all plates and welds in the beltline region of those units, respectively. All transition temperature shifts were calculated to be less than 100°F. Therefore, the minimum number of surveillance capsules to be withdrawn is three, in accordance with ASTM E 185-82. Per ASTM E 185-82, the withdrawal of a capsule is scheduled for the vessel refueling outage nearest to the calculated EFPY established for the particular surveillance capsule withdrawal.

References 1 and 2 indicate that the first two capsules removed from the CPNPP Units 1 and 2 RVs met the intent of ASTM E 185-82. CPNPP Unit 1 has already removed its third surveillance capsule - Capsule X. Capsule X still satisfies the criteria for the third capsule withdrawn in a three-capsule withdrawal schedule (see Note E in Table 1 of ASTM E 183-82), since the capsule fluence, 3.24×10^{19} n/cm^2, E > 1.0 MeV, is greater than 2.23×10^{19} n/cm^2, E > 1.0 MeV (36 EFPY end-of-life (EOL) vessel fluence) and less than 4.46×10^{19} n/cm^2, E > 1.0 MeV (two-times the EOL vessel fluence). Therefore, the CPNPP Unit 1 withdrawal schedule also meets the requirement of ASTM E 185-82 for "third capsule" considering the projected neutron fluence after the SPU for a 36 EFPY EOL. CPNPP Unit 2 has not yet removed its third surveillance capsule.

As presented in LR Section 1.1, Nuclear Steam Supply System Parameters, the RV inlet temperature is maintained above 525°F and below 590°F. Therefore, the equations and results remain valid without adjustments for temperature effects.

Surveillance capsule withdrawal schedules exist in References 1 and 2 that meet the intent of ASTM E 185-82 and 10 CFR Part 50, Appendix H. Having these withdrawal schedules satisfies 10 CFR Part 50.60, GDC-14, and -31.

2.1.1.3 Conclusions

Luminant Power has reviewed the evaluation of the effects of the proposed SPU on the RV surveillance withdrawal schedule and concludes that Luminant Power has adequately addressed changes in neutron fluence and their effects on the schedules for CPNPP Units 1 and 2. Luminant Power further concludes that the RV capsule withdrawal schedules are appropriate to ensure that the material surveillance programs will continue to meet the requirements of 10 CFR Part 50, Appendix H, and 10 CFR 50.60, and will provide Luminant Power with information to ensure continued compliance with GDC-14 and -31 in this respect following implementation of the proposed SPU. Therefore, Luminant Power finds the proposed SPU acceptable with respect to the RV material surveillance programs.

2.1.1.4 References

1. WCAP-16610, “Analysis of Capsule X from the TXU Energy Comanche Peak Unit 1 Reactor Vessel Radiation Surveillance Program,” September 2006.

2. WCAP-16277, “Analysis of Capsule X from the TXU Energy Comanche Peak Unit 2 Reactor Vessel Radiation Surveillance Program,” September 2004.

3. Regulatory Guide RG-1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, March 2001.

4. WCAP-14040, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," May 2004.

5. RSICC Computer Code Collection CCC-650, "DOORS 3.2, One-, Two-, and Three-Dimensional Discrete Ordinates Neutron/Photon Transport Code System," April 1998.

6. RSIC Data Library Collection DLC-185, "BUGLE-96, Coupled 47 Neutron, 20 Gamma Ray Group Cross-Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications," March 1996.

Table 2.1.1-1

Maximum Calculated Neutron Exposure of the Reactor Vessel Beltline Materials at the Clad/Base Metal Interface (n/cm^2, ($E > 1.0$ MeV))

Projected EFPY	Azimuthal Location				
	0°	15°	21°	30°	45°
CPNPP Unit 1					
15.9[(1)]	6.26E+18	8.99E+18	9.90E+18	9.21E+18	9.13E+18
20.0	8.01E+18	1.14E+19	1.24E+19	1.13E+19	1.09E+19
25.0	1.02E+19	1.44E+19	1.55E+19	1.38E+19	1.31E+19
32.0	1.32E+19	1.86E+19	1.98E+19	1.74E+19	1.61E+19
36.0	1.49E+19	2.09E+19	2.23E+19	1.94E+19	1.78E+19
40.0	1.66E+19	2.33E+19	2.47E+19	2.14E+19	1.96E+19
48.0	2.00E+19	2.81E+19	2.97E+19	2.55E+19	2.30E+19
54.0	2.26E+19	3.17E+19	3.33E+19	2.85E+19	2.56E+19
CPNPP Unit 2					
14.5[(2)]	6.14E+18	8.51E+18	9.20E+18	8.37E+18	8.31E+18
20.0	8.51E+18	1.18E+19	1.26E+19	1.12E+19	1.07E+19
25.0	1.07E+19	1.48E+19	1.57E+19	1.37E+19	1.29E+19
32.0	1.37E+19	1.89E+19	2.00E+19	1.73E+19	1.59E+19
36.0	1.54E+19	2.13E+19	2.25E+19	1.93E+19	1.77E+19
40.0	1.71E+19	2.37E+19	2.49E+19	2.13E+19	1.94E+19
48.0	2.05E+19	2.85E+19	2.98E+19	2.54E+19	2.29E+19
54.0	2.31E+19	3.20E+19	3.35E+19	2.84E+19	2.55E+19

Notes:

1. Projected End of Cycle 13, uprate to 3,612 MWt is assumed to occur at the onset of Cycle 14.
2. Projected End of Cycle 11, uprate to 3,612 MWt is assumed to occur at the onset of Cycle 12.

Table 2.1.1-2

Recommended Surveillance Capsule Withdrawal Schedule for CPNPP Unit 1

Capsule	Capsule Location	Lead Factor(1)	Withdrawal EFPY(2)	Fluence (n/cm^2)(1)
U	58.5°	4.01	0.91	3.18×10^{18}
Y	241.0°	3.86	6.24	1.49×10^{19}
X	238.5°	3.97	13.10	3.24×10^{19}(3)
Z	301.5°	3.93	Standby(4)	(4)
W	121.5°	3.99	10.42(5)	2.23×10^{19}
V	61.0°	3.74	10.42(5)	2.07×10^{19}

Notes:

1. Updated in Capsule X dosimetry analysis, see Reference 1.
2. EFPYs from plant startup.
3. This fluence is greater than one-times and less than two-times the peak EOL (@ 36 EFPY) vessel fluence.
4. If license renewal is obtained, this capsule should be withdrawn anytime after 13.5 EFPY, which is when the capsule fluence would exceed one-times the peak end of license renewal (EOLR) (52 EFPY) vessel fluence, not to exceed 26.7 EFPY, which is when the fluence on the capsule would exceed two-times the EOLR (52 EFPY) vessel fluence. If license renewal is never obtained, then this capsule should still be removed prior to 26.7 EFPY and placed in storage. See Note "5".
5. These capsules were withdrawn after 10.42 EFPY and placed into storage. Once all capsules are removed alternative fluence measuring capabilities must be in place.

Table 2.1.1-3

Recommended Surveillance Capsule Withdrawal Schedule for CPNPP Unit 2

Capsule	Capsule Location	Lead Factor(1)	Withdrawal EFPY(2)	Fluence (n/cm^2)(1)
U	58.5°	3.93	0.91	3.15×10^{18}
X	238.5°	4.15	8.83	2.20×10^{19}
W	121.5°	4.11	11.7(3)	- -
Z	301.5°	4.11	Standby(4)	(4)
V	61.0°	3.87	Standby(5)	(5)
Y	241.0°	3.87	Standby(5)	(5)

Notes:

1. Updated in Capsule X dosimetry analysis, see Reference 2.
2. EFPY from plant startup.
3. Surveillance Capsule W withdrawal EFPY is based on the recommendations provided in the surveillance capsule X Analysis in Reference 2.
4. If license renewal is obtained, this capsule should be withdrawn anytime after 13.6 EFPY, which is when the capsule fluence would exceed one-times the peak EOLR (54 EFPY) vessel fluence, not to exceed 26.6 EFPY, which is when the fluence on the capsule would exceed two-times the EOLR (54 EFPY) vessel fluence. If license renewal is never obtained, then this capsule should still be removed prior to 26.6 EFPY and placed in storage. See Note "5".
5. These capsules should be withdrawn at least one outage prior to 28.2 EFPY, which is when the fluence on these capsules would exceed two-times the EOLR (54 EFPY) vessel fluence. Once all capsules are removed, alternative fluence measuring capabilities must be in place.

Table 2.1.1-4 Summary of the CPNPP Units 1 and 2 Beltline Material Chemistry and Chemistry Factors Based on Regulatory Guide 1.99, Revision 2				
Beltline Materials	**Wt. % Cu**	**Wt. % Ni**	**Position 1.1 CF**	**Position 2.1 CF**
CPNPP Unit 1				
Intermediate Shell Plate R-1107-1	0.07	0.62	44.0°F	--
Intermediate Shell Plate R-1107-2	0.07	0.67	44.0°F	--
Intermediate Shell Plate R-1107-3	0.06	0.65	37.0°F	--
Lower Shell Plate R-1108-1	0.08	0.65	51.0°F	--
Lower Shell Plate R-1108-2	0.06	0.60	37.0°F	24.3°F
Lower Shell Plate R-1108-3	0.08	0.65	51.0°F	--
Beltline Region Weld Metal	0.045	0.20	46.0°F	15.7°F
CPNPP Unit 2				
Intermediate Shell Plate R-3807-1	0.06	0.64	37°F	--
Intermediate Shell Plate R-3807-2	0.06	0.64	37°F	21.6°F
Intermediate Shell Plate R-3807-3	0.05	0.60	31°F	--
Lower Shell Plate R-3816-1	0.05	0.59	31°F	--
Lower Shell Plate R-3816-2	0.03	0.65	20°F	--
Lower Shell Plate R-3816-3	0.04	0.63	26°F	--
Intermediate and Lower Shell Longitudinal Welds	0.046	0.059	31.5°F	32.8°F
Intermediate to Lower Shell Girth Weld	0.046	0.059	31.5°F	32.8°F

Table 2.1.1-5 Summary of the CPNPP Units 1 and 2 ΔRT_{NDT} Determination					
Material	**RG 1.99 R2 Method**	**CF (°F)**	**Fluence (10^{19} n/cm^2)**	**FF**	**ΔRT_{NDT} (°F)**
CPNPP Unit 1					
Intermediate Shell Plate R-1107-1	Position 1.1	44.0	2.23	1.2173	53.56
Intermediate Shell Plate R-1107-2	Position 1.1	44.0	2.23	1.2173	53.56
Intermediate Shell Plate R-1107-3	Position 1.1	37.0	2.23	1.2173	45.04
Lower Shell Plate R-1108-1	Position 1.1	51.0	2.23	1.2173	62.08
Lower Shell Plate R-1108-2	Position 1.1	37.0	2.23	1.2173	45.04
	Position 2.1	24.3	2.23	1.2173	29.58
Lower Shell Plate R-1108-3	Position 1.1	51.0	2.23	1.2173	62.08
Beltline Region Weld Metal	Position 1.1	46.0	2.23	1.2173	56.00
	Position 2.1	15.7	2.23	1.2173	19.11
CPNPP Unit 2					
Intermediate Shell Plate R-3807-1	Position 1.1	37.0	2.25	1.2196	45.12
Intermediate Shell Plate R-3807-2	Position 1.1	37.0	2.25	1.2196	45.12
	Position 2.1	21.6	2.25	1.2196	26.34
Intermediate Shell Plate R-3807-3	Position 1.1	31.0	2.25	1.2196	37.81
Lower Shell Plate R-3816-1	Position 1.1	31.0	2.25	1.2196	37.81
Lower Shell Plate R-3816-2	Position 1.1	20.0	2.25	1.2196	24.39
Lower Shell Plate R-3816-3	Position 1.1	26.0	2.25	1.2196	31.71
Intermediate and Lower Shell Longitudinal Welds	Position 1.1	31.5	2.25	1.2196	38.42
	Position 2.1	32.8	2.25	1.2196	40.00
Intermediate to Lower Shell Girth Weld	Position 1.1	31.5	2.25	1.2196	38.42
	Position 2.1	32.8	2.25	1.2196	40.00

2.1.2 Pressure-Temperature Limits and Upper Shelf Energy

2.1.2.1 Regulatory Evaluation

Pressure-temperature (P-T) limits are established to ensure the structural integrity of the ferritic components of the reactor coolant pressure boundary (RCPB) during any condition of normal operation, including anticipated operation occurrences and hydrostatic tests. Luminant Power's review of P-T limits covered the P-T limits methodology and the calculations for the number of effective full-power years (EFPYs) specified for the stretch power uprate (SPU), considering neutron embrittlement effects and using linear elastic fracture mechanics.

The acceptance criteria are based on:

- General Design Criterion (GDC)-14, insofar as it requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture.

- GDC-31, insofar as it requires that the RCPB be designed with margin sufficient to assure that, under specific conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized.

- 10 CFR Part 50, Appendix G, which specifies fracture toughness requirements for ferritic components of the RCPB.

- 10 CFR Part 50.60, which requires compliance with the requirements of 10 CFR Part 50, Appendix G.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the CPNPP reactor protection, engineered safety feature actuation, and control systems for adequacy regarding conformance to:

- GDC-14, Reactor Coolant Pressure Boundary, is described in FSAR Section 3.1.2.5.

  The reactor coolant system (RCS) pressure boundary is designed to accommodate the system pressures and temperatures attained under the expected modes of plant operation, including anticipated transients, and to maintain the stresses within applicable stress limits (FSAR Section 5.2). Also, RCPB material and selection and fabrication techniques ensure a low probability of gross rupture of significant leakage.

In addition to the loads imposed on the system under normal operating conditions, consideration is also given to abnormal loading conditions, such as seismic and pipe rupture, which are discussed in FSAR Sections 3.6 and 3.7.

The RCS boundary has provisions for inspection, testing, and surveillance of critical areas to assess the structural and leak-tight integrity (FSAR Section 5.2). For the reactor vessel, a material surveillance program conforming to applicable codes is provided (FSAR Section 5.3).

- GDC-31, Fracture Prevention of Reactor Coolant Pressure boundary, is described in FSAR Section 3.1.4.2.

  Close control is maintained over material selection and fabrication for the RCS to ensure that the boundary behaves in a nonbrittle manner. Those RCS materials that are exposed to the coolant are corrosion resistant, stainless steel or Inconel. The reference temperature (RT_{NDT}) of the reactor vessel structural steel is established by the Charpy V-notch and drop weight tests in accordance with 10 CFR Part 50, Appendix G.

2.1.2.2 Technical Evaluation

2.1.2.2.1 Introduction

Reactor vessel integrity is impacted by any change in plant parameters that affect neutron fluence levels or temperature/pressure transients. The changes in neutron fluence resulting from the SPU have been evaluated to determine the impact on reactor vessel integrity. The assessment presented herein focuses on the CPNPP Units 1 and 2 P-T limits developed in WCAP-16346 (Reference 1) and the projected values of upper shelf energy at end-of-life (EOL) (36 EFPY). In this section, 36 EFPY vessel surface fluence values under SPU conditions are compared against those used to determine adjusted reference temperatures (ARTs) (RT_{NDT}) in Reference 1 for development of the CPNPP Units' P-T limits. The projected decrease in upper shelf energy (USE) due to irradiation embrittlement based on uprated fluence values is evaluated to ensure adequate margin in USE at EOL (36 EFPY).

2.1.2.2.2 Input Parameters, Assumptions, and Acceptance Criteria

SPU Fluence Projections

Neutron fluence projections as discussed in Licensing Report (LR) subsection 2.1.1.2.2 considering SPU conditions are presented in Table 2.1.2-1 for the beltline materials in both CPNPP Units 1 and 2. The calculated fluence projections used in the SPU evaluation complied with Regulatory Guide 1.190. As these calculations were performed on a plant-by-plant basis, there was no generic topical report for the approved method. The methodology used was that of Regulatory Guide 1.190.

Inlet Temperature

As presented in LR Section 1.1, Nuclear Steam Supply System Parameters, the SPU full-power reactor vessel inlet temperature range is 542.2° to 558.0°F.

Chemistry

Chemistry of the plates and welds, specifically the weight percent copper, was used along with neutron fluence to determine the predicted decrease in USE at end of license (36 EFPY). The weight percent copper for all the beltline and extended beltline materials is presented in Table 2.1.2-2.

Upper Shelf Energy

The initial USE values for each plate and weld in the beltline region are used in determining the projected USE values at 36 EFPYs. These initial USE values are presented in Table 2.1.2-2.

Pressure-Temperature Limits

The P-T limit curves are contained in Reference 1, as determined for a 36 EFPY EOL. CPNPP uses common P-T limit curves for both units. Reference 1 indicated that the Unit 1 1/4T position adjusted reference temperature (ART) was limiting, based on a projected neutron fluence of 2.45×10^{19} n/cm^2 (E>1.0 MeV). Reactor vessel integrity evaluations provided in Reference 1 form the basis for ART values of the Technical Specification P-T limits.

Acceptance Criteria

For P-T limit curves, the acceptance criteria are that Luminant Power has NRC-approved P-T limits developed in accordance with 10 CFR Part 50, Appendix G, and that the applicable EFPY of those P-T limit curves after implementation of the SPU does not invalidate the term of applicability.

For USE at SPU conditions, 36 EFPY values for all reactor beltline materials must meet the requirements of 10 CFR Part 50, Appendix G. This states that the USE must be maintained above 50 ft-lbs, otherwise an equivalent margins analysis must be performed to demonstrate that the vessel has adequate margin of safety.

The acceptance criteria for the reactor vessel inlet temperature are provided in NRC Regulatory Guide 1.99, Revision 2, Radiation Embrittlement of Reactor Vessel Materials. This states that "The procedures are valid for a nominal irradiation temperature of 550°F. Irradiation below 525°F should be considered to produce greater embrittlement, and irradiation above 590°F may be considered to produce less embrittlement." Therefore, the reactor vessel inlet temperature must be greater than 525°F and less than 590°F for the equations and methodology of Regulatory Guide 1.99, Revision 2 to remain valid.

2.1.2.2.3 Description of Analyses and Evaluations

If the post-power-uprate reactor vessel fluence projections at 36 EFPY exceed those in the analysis of record, then a new applicability date of the current P-T limit curves would need to be calculated. If the post-power-uprate reactor vessel fluence projections for both units are lower than the 36 EFPY neutron fluence values utilized in the analysis of record then, conservatively, no change to the applicability date is required.

The evaluation to assess the impact of the SPU on USE requires that the percentage decrease in USE be determined in accordance with Regulatory Guide 1.99, Revision 2 for each plate and weld in the vessel beltline. Percentage decreases in USE, from the initial unirradiated USE, can be predicted as a function of neutron fluence for plates and welds of known copper content. Fluence values used to determine USE decreases are those at the 1/4 vessel thickness, using the fluence attenuation formula provided in Regulatory Guide 1.99, Revision 2. Values for USE at EOL (36 EFPY) are then evaluated against the acceptance criteria of 50 ft-lbs in 10 CFR Part 50, Appendix G.

Evaluation of the proposed CPNPP SPU also includes a review of the reactor vessel inlet temperature to verify that it complies with Regulatory Guide 1.99, Revision 2, which provides the embrittlement correlations used to calculate changes to ART (for determination of P-T Limit Curves) and USE as a function of neutron fluence.

2.1.2.2.4 Pressure-Temperature Limits and Upper Shelf Energy Results

Reactor vessel fluence projections were generated for SPU conditions following the guidance of Regulatory Guide 1.190 (see Table 2.1.2-1).

At 36 EFPY, the maximum projected fluence on the reactor vessel beltline, accounting for SPU conditions, would be 2.23×10^{19} n/cm^2 (E > 1.0 MeV) for CPNPP Unit 1 and 2.25×10^{19} n/cm^2 (E > 1.0 MeV) for CPNPP Unit 2. Luminant Power has developed P-T limit curves applicable to 36 EFPY for both units based on a neutron fluence exposure of 2.45×10^{19} n/cm^2 (E > 1.0 MeV) on the limiting material in CPNPP Unit 1 (see Reference 1). The updated neutron fluence projection for CPNPP Unit 1 is approximately 10-percent lower than the value used in the Reference 1 analysis used to construct the P-T limit curves. Note that the updated neutron fluence projection at 36 EFPY is also less for CPNPP Unit 2 than the value evaluated in Reference 1. The initial RT_{NDT} and chemistry factor values for the vessel materials are unchanged as a result of the SPU. ART values calculated with a lower fluence considering SPU conditions would, therefore, be correspondingly lower in magnitude. Therefore, no changes to the date of applicability for the P-T limit curves are required.

Neutron fluence values at 36 EFPY for the 1/4T vessel thickness location were used to predict the decrease in USE for materials in the reactor vessel beltline. Table 2.1.2-2 provides the copper chemistry and initial USE for those materials. Copper chemistry and 1/4T neutron fluence were used in accordance with Regulatory Guide 1.99, Revision 2 to predict the percentage decrease in USE at EOL. Surveillance capsule test data is also used in accordance with Position 2.2 of Regulatory Guide 1.99, Revision 2 for predicting the decrease

in USE. Reduced surveillance data is plotted in Figure 2 of the Guide, as replicated in Figures 2.1.2-1 and 2.1.2-2. Figure 2 of Regulatory Guide 1.99, Revision 2 does not show USE decreases below 2 percent. For CPNPP Unit 2, where no decrease in USE has thus far been measured for surveillance plate specimens (Capsules U and X, both longitudinal and transverse orientations), there are no surveillance points plotted in Figure 2.1.2-2. Only Position 1.2 of Regulatory Guide 1.99, Revision 2 was used to predict the drop in USE for CPNPP Unit 2, Intermediate Shell Plate R-3807-2. The USE predictions for all vessel materials are provided in Table 2.1.2-3, which demonstrates that all plates and welds in both units are predicted to have USE values that remain above 50 ft-lbs at EOL.

As presented in LR Section 1.1, Nuclear Steam Supply System Parameters, the reactor vessel inlet temperature is maintained above 525°F and below 590°F. Therefore, the equations and results remain valid without adjustments for temperature effects.

An NRC-approved set of P-T limit curves exists in the CPNPP Technical Specifications which satisfies the requirements of 10 CFR Part 50, Appendix G for a 36 EFPY term of applicability, with consideration of the SPU neutron fluence exposure. Additionally, the SPU fluence projections were shown not to reduce the level of USE for any plate or weld in the beltline to below 50 ft-lbs at EOL in accordance with 10 CFR Part 50, Appendix G.

2.1.2.3 Conclusions

Luminant Power has reviewed the evaluation of the effects of the proposed SPU on the P-T limits for CPNPP Units 1 and 2. The evaluation has adequately addressed changes in neutron fluence and their effects on the P-T limits. Luminant Power further concludes that all plates and welds in the CPNPP Units 1 and 2 reactor vessel beltline have projected values for USE above 50 ft-lbs at EOL (36 EFPY). Based on this evaluation, the current P-T limits will continue to meet the requirements of 10 CFR Part 50, Appendix G, and 10 CFR 50.60 and will enable Luminant Power to comply with GDC-14 and GDC-31 in this respect following implementation of the proposed SPU. Therefore, Luminant Power finds the proposed SPU acceptable with respect to the current P-T limits and USE screening criteria.

2.1.2.4 References

1. WCAP-16346, Revision 0, "Comanche Peak Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation," October 2004.

2. WCAP-16610, Rev. 0, "Analysis of Capsule X from the TXU Energy Comanche Peak Unit 1 Reactor Vessel Radiation Surveillance Program," September 2006.

3. WCAP-16277, Rev. 0, "Analysis of Capsule X from the TXU Energy Comanche Peak Unit 2 Reactor Vessel Radiation Surveillance Program," September 2004.

Table 2.1.2-1 Maximum Calculated Neutron Exposure of the Reactor Vessel Beltline Materials at the Clad/Base Metal Interface (n/cm^2, (E > 1.0 MeV))					
Projected EFPY	Azimuthal Location				
	0°	15°	21°	30°	45°
CPNPP Unit 1					
15.9 (1)	6.26E+18	8.99E+18	9.90E+18	9.21E+18	9.13E+18
20.0	8.01E+18	1.14E+19	1.24E+19	1.13E+19	1.09E+19
25.0	1.02E+19	1.44E+19	1.55E+19	1.38E+19	1.31E+19
32.0	1.32E+19	1.86E+19	1.98E+19	1.74E+19	1.61E+19
36.0	1.49E+19	2.09E+19	2.23E+19	1.94E+19	1.78E+19
40.0	1.66E+19	2.33E+19	2.47E+19	2.14E+19	1.96E+19
48.0	2.00E+19	2.81E+19	2.97E+19	2.55E+19	2.30E+19
54.0	2.26E+19	3.17E+19	3.33E+19	2.85E+19	2.56E+19
CPNPP Unit 2					
14.5 (2)	6.14E+18	8.51E+18	9.20E+18	8.37E+18	8.31E+18
20.0	8.51E+18	1.18E+19	1.26E+19	1.12E+19	1.07E+19
25.0	1.07E+19	1.48E+19	1.57E+19	1.37E+19	1.29E+19
32.0	1.37E+19	1.89E+19	2.00E+19	1.73E+19	1.59E+19
36.0	1.54E+19	2.13E+19	2.25E+19	1.93E+19	1.77E+19
40.0	1.71E+19	2.37E+19	2.49E+19	2.13E+19	1.94E+19
48.0	2.05E+19	2.85E+19	2.98E+19	2.54E+19	2.29E+19
54.0	2.31E+19	3.20E+19	3.35E+19	2.84E+19	2.55E+19

Notes:

1. Projected EOL 13, uprate to 3,612 MWt is assumed to occur at the onset of Cycle 14.
2. Projected EOL 11, uprate to 3,612 MWt is assumed to occur at the onset of Cycle 12.

Table 2.1.2-2

CPNPP Units 1 and 2 Beltline Region Materials Properties

CPNPP Unit 1		
Reactor Vessel Beltline Materials	**Wt. % Cu**	**Initial USE**
Intermediate Shell Plate R-1107-1	0.07	94 ft-lb
Intermediate Shell Plate R-1107-2	0.07	103 ft-lb
Intermediate Shell Plate R-1107-3	0.06	88 ft-lb
Lower Shell Plate R-1108-1	0.08	85 ft-lb
Lower Shell Plate R-1108-2	0.06	78 ft-lb
Lower Shell Plate R-1108-3	0.08	98 ft-lb
Beltline Region Weld Metal	0.045	133 ft-lb
CPNPP Unit 2		
Reactor Vessel Beltline Materials	**Wt. % Cu**	**Initial USE**
Intermediate Shell Plate R-3807-1	0.06	108 ft-lb
Intermediate Shell Plate R-3807-2	0.06	101 ft-lb
Intermediate Shell Plate R-3807-3	0.05	105 ft-lb
Lower Shell Plate R-3816-1	0.05	107 ft-lb
Lower Shell Plate R-3816-2	0.03	106 ft-lb
Lower Shell Plate R-3816-3	0.04	108 ft-lb
Intermediate and Lower Shell Longitudinal Welds	0.046	172 ft-lb
Intermediate to Lower Shell Girth Weld	0.046	96 ft-lb

Table 2.1.2-3

USE Prediction Calculations at 36 EFPY for the CPNPP Beltline Region Materials

CPNPP Unit 1				
Material	**1/4T EOL Fluence (10^{19} n/cm^2)**	**Unirradiated USE (ft-lb)**	**Projected USE Decrease (%)**	**Projected EOL USE (ft-lb)**
Intermediate Shell Plate R-1107-1	1.3287	94	20.5[(1)]	74.7
Intermediate Shell Plate R-1107-2	1.3287	103	20.5[(1)]	81.9
Intermediate Shell Plate R-1107-3	1.3287	88	20.5[(1)]	70.0
Lower Shell Plate R-1108-1	1.3287	85	20.5[(1)]	67.6
Lower Shell Plate R-1108-2	1.3287	78	8.0[(2)]	71.8
Lower Shell Plate R-1108-3	1.3287	98	20.5[(1)]	77.9
Beltline Region Weld Metal	1.3287	133	4.8[(2)]	126.6

CPNPP Unit 2				
Material	**1/4T EOL Fluence (10^{19} n/cm^2)**	**Unirradiated USE (ft-lb)**	**Projected USE Decrease (%)**	**Projected EOL USE (ft-lb)**
Intermediate Shell Plate R-3807-1	1.3406	108	20.5[(1)]	85.9
Intermediate Shell Plate R-3807-2	1.3406	101	20.5[(1)]	80.3
Intermediate Shell Plate R-3807-3	1.3406	105	20.5[(1)]	83.5
Lower Shell Plate R-3816-1	1.3406	107	20.5[(1)]	85.1
Lower Shell Plate R-3816-2	1.3406	106	20.5[(1)]	84.3
Lower Shell Plate R-3816-3	1.3406	108	20.5[(1)]	85.9
Intermediate and Lower Shell Longitudinal Welds	1.3406	172	14.0[(2)]	147.9
Intermediate to Lower Shell Girth Weld	1.3406	96	14.0[(2)]	82.6

Notes:

1. Projected USE decrease in accordance with Regulatory Guide 1.99, Revision 2, Position 1.2.
2. Using the most recent surveillance capsule analysis data from References 2 and 3 in accordance with Position 2.2 of Regulatory Guide 1.99, Revision 2 (see Figures 2.1.2-1 and 2.1.2-2). The limiting measured decreases in USE for the Unit 1 surveillance materials were 6% for the weld metal specimens from Capsule X and 10% for the longitudinally-oriented plate specimens from Capsule X. The limiting measured decreases in USE for the Unit 2 surveillance materials were 10% for the weld metal specimens from Capsule U. No decreases were measured for the Unit 2 surveillance plate specimens, so the Position 1.2 USE prediction was conservatively used to demonstrate compliance with regulatory requirements.

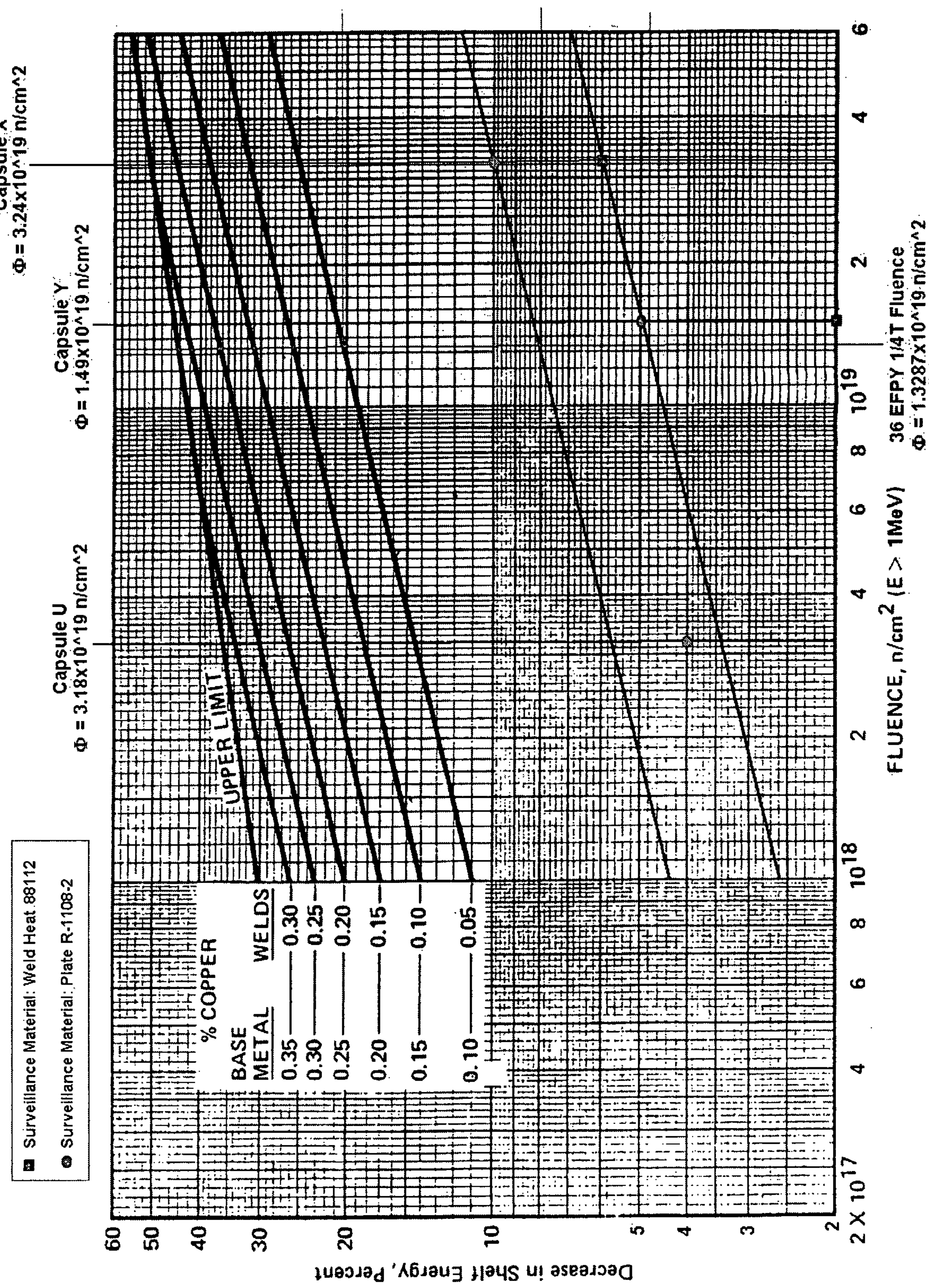

Figure 2.1.2-1 **Regulatory Guide 1.99, Revision 2, Predicted Decrease in Upper Shelf Energy as a Function of Copper and Fluence for CPNPP Unit 1, Including Surveillance Data**

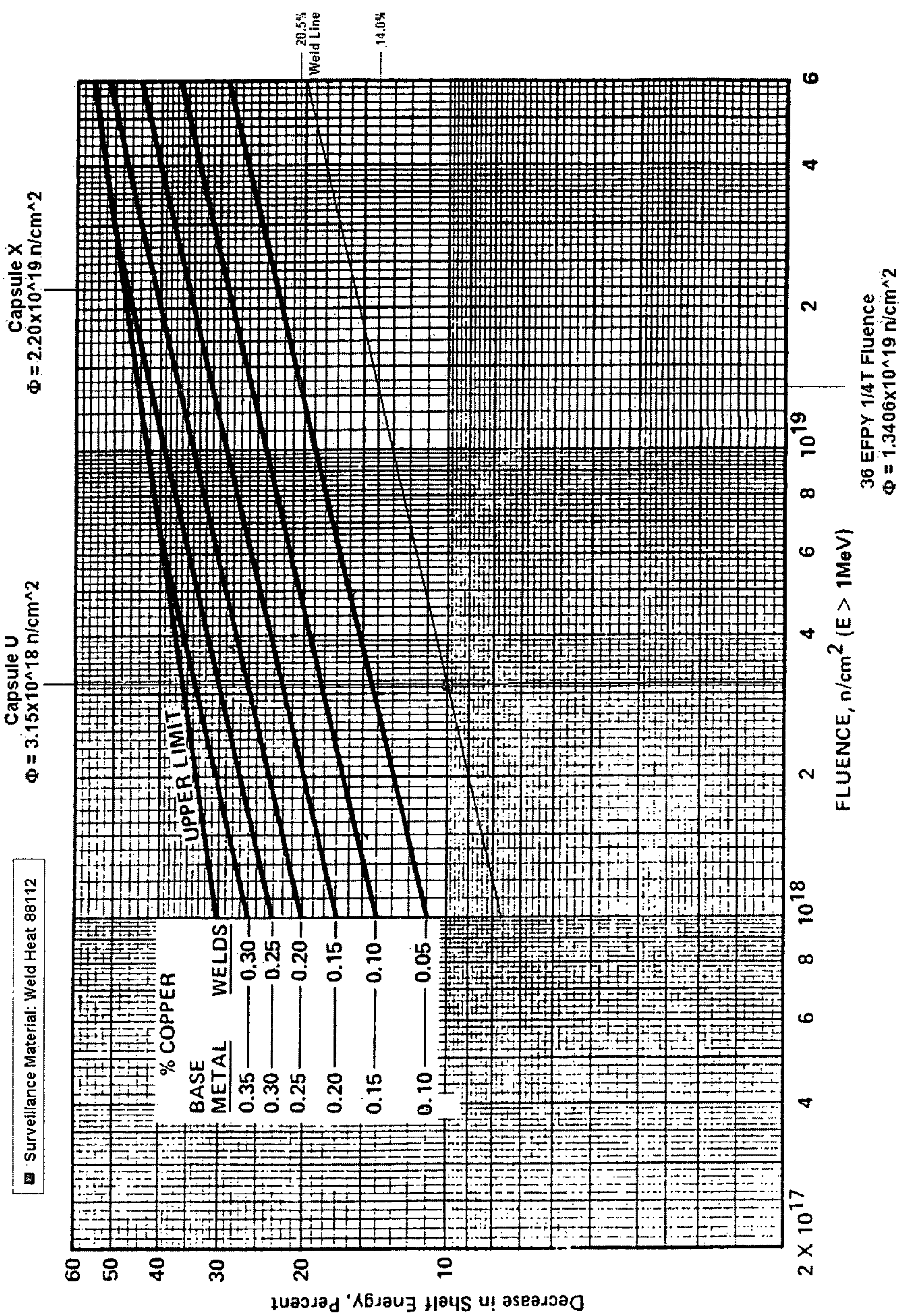

Figure 2.1.2-2 **Regulatory Guide 1.99, Revision 2, Predicted Decrease in Upper Shelf Energy as a Function of Copper and Fluence for CPNPP Unit 2, Including Surveillance Data**

2.1.3 Pressurized Thermal Shock

2.1.3.1 Regulatory Evaluation

The pressurized thermal shock (PTS) evaluation provides a means for assessing the susceptibility of the reactor vessel beltline materials to PTS events to ensure that adequate fracture toughness is provided for supporting reactor operations. Luminant Power reviewed the current license basis for the PTS methodology and the calculations for the referenced temperature (RT_{PTS}) at the expiration of license, considering neutron embrittlement effects.

The acceptance criteria are based on:

- General Design Criterion (GDC)-14, insofar as it requires that the reactor coolant pressure boundary (RCPB) be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating fracture, and of gross rupture.

- GDC-31, insofar as it requires that the RCPB be designed with margin sufficient to ensure that, under specific conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized.

- 10CFR Part 50.61, insofar as it sets fracture toughness criteria for protection against PTS events.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the CPNPP reactor protection, engineered safety feature actuation, and control systems for adequacy regarding conformance to:

- GDC-14, Reactor Coolant Pressure Boundary, is described in FSAR Section 3.1.2.5.

  The reactor coolant system (RCS) pressure boundary is designed to accommodate the system pressures and temperatures attained under the expected modes of plant operation, including anticipated transients, and to maintain the stresses within applicable stress limits (FSAR Section 5.2). Also, RCPB material and selection and fabrication techniques ensure a low probability of gross rupture of significant leakage.

  In addition to the loads imposed on the system under normal operating conditions, consideration is also given to abnormal loading conditions, such as seismic and pipe rupture, which are discussed in FSAR Sections 3.6 and 3.7.

The RCS boundary has provisions for inspection, testing, and surveillance of critical areas to assess the structural and leak-tight integrity (FSAR Section 5.2). For the reactor vessel, a material surveillance program conforming to applicable codes is provided (FSAR Section 5.3).

- GDC-31, Fracture Prevention of Reactor Coolant Pressure boundary, is described in FSAR Section 3.1.4.2.

  Close control is maintained over material selection and fabrication for the RCS to ensure that the boundary behaves in a nonbrittle manner. Those RCS materials that are exposed to the coolant are corrosion resistant, stainless steel, or Inconel. The reference temperature (RT_{NDT}) of the reactor vessel structural steel is established by the Charpy V-notch and drop weight tests in accordance with 10 CFR Part 50, Appendix G.

2.1.3.2 Technical Evaluation

2.1.3.2.1 Introduction

Reactor vessel integrity is impacted by any change in plant parameters that affect neutron fluence levels or temperature/pressure transients. The changes in neutron fluence resulting from the stretch power uprate (SPU) have been evaluated to determine the impact on reactor vessel integrity. The assessment presented herein focuses on the CPNPP Units 1 and 2 reference temperatures for pressurized thermal shock at end-of-life (EOL) (36 effective full-power years (EFPYs)).

2.1.3.2.2 Input Parameters, Assumptions, and Acceptance Criteria

SPU Fluence Projections

Neutron fluence projections (as discussed in Licensing Report (LR) subsection 2.1.1.2.2) considering SPU conditions are presented in Table 2.1.3-1 for the beltline materials in both CPNPP Units 1 and 2. The calculated fluence projections used in the SPU evaluation complied with Regulatory Guide 1.190. As these calculations were performed on a plant-by-plant basis, there was no generic topical report for the approved method. The methodology used was that of Regulatory Guide 1.190.

Inlet Temperature

As presented in LR Section 1.1, Nuclear Steam Supply System Parameters, the SPU full-power reactor vessel inlet temperature range is 542.2° to 558.0°F.

Chemistry Factor Values

The chemistry factors (CFs), along with the fluence factors (FFs), are used to determine the shift in reference temperature, ΔRT_{NDT}. The CF is a function of the copper and nickel content, and is determined in accordance with Tables 1 and 2 of 10 CFR Part 50.61. In accordance with

10 CFR Part 50.61 Section (c)(2), those plate and weld materials that are part of a plant-specific surveillance program, must have material-specific CFs calculated and incorporated into the determination of the RT_{NDT} if the surveillance data are deemed credible. The CFs used in this evaluation are presented in Table 2.1.3-2, along with the best-estimate copper and nickel chemistry used to calculate the CF values from Tables 1 and 2 of 10 CFR Part 50.61. For clarity and consistency with Regulatory Guide 1.99, Revision 2 CFs calculated based on chemistry are referred to as Position 1.1 and CFs calculated based on surveillance data are referred to as Position 2.1.

Initial Reference Temperature, Nil-Ductility Temperature (RT_{NDT})

The unirradiated material reference temperatures (RT_{NDT}) for the beltline materials were determined from laboratory testing as part of the development of the plant-specific radiation surveillance programs. References 1 and 2 identify the reactor vessel unirradiated RT_{NDT} values for CPNPP Units 1 and 2, respectively. These values are identified in Table 2.1.3-3 under the column $RT_{NDT(U)}$.

Acceptance Criteria

Criteria for acceptance of reference temperature predictions for PTS are provided in 10 CFR Part 50.61. The RT_{PTS} values must not exceed 270°F for plates, forgings, and axial welds, and below 300°F for circumferential welds.

The acceptance criteria for the reactor vessel inlet temperature are provided in U.S. NRC Regulatory Guide 1.99, Revision 2, Radiation Embrittlement of Reactor Vessel Materials. This states that "The procedures are valid for a nominal irradiation temperature of 550°F. Irradiation below 525°F should be considered to produce greater embrittlement, and irradiation above 590°F may be considered to produce less embrittlement." Therefore, the reactor vessel inlet temperature must be greater than 525°F and less than 590°F for the equations and methodology of Regulatory Guide 1.99, Revision 2 to remain valid.

2.1.3.2.3 Description of Analyses and Evaluations

The limiting condition on reactor vessel integrity known as PTS can occur during a severe system transient such as a loss-of-coolant accident (LOCA) or a steam line break. Such transients can challenge the integrity of a reactor vessel under the following conditions:

- Severe overcooling of the inside surface of the vessel wall followed by high repressurization
- Significant degradation of vessel material toughness caused by radiation embrittlement
- Presence of a critical-size defect in the vessel wall

The PTS concern arises if one of these transients should act on the beltline region of a reactor vessel where a reduced fracture resistance exists because of neutron irradiation. Such an

event could cause the propagation of flaws postulated to exist near the inner wall surface, thereby potentially affecting the integrity of the vessel.

In 1985, the NRC issued a formal ruling on PTS. It established screening criteria on pressurized water reactor (PWR) vessel embrittlement as measured by the RT_{PTS}. RT_{PTS} screening criteria values were set (using conservative fracture mechanics analysis techniques) for beltline axial welds, plates, and beltline circumferential weld seams for EOL plant operation. All PWR vessels in the U.S. have been required to evaluate vessel embrittlement in accordance with the criteria through EOL.

The NRC subsequently amended its regulations for light water reactors (LWRs) changing the procedure for calculating radiation embrittlement. The revised PTS rule was published in the Federal Register, December 19, 1995, with an effective date of January 18, 1996. This amendment made the procedure for calculating RT_{PTS} values consistent with the methods given in Regulatory Guide 1.99, Revision 2.

The PTS rule establishes the following requirements for all domestic operating PWRs:

- For each PWR that has had an operating license issued, the Licensee will have projected values of RT_{PTS} accepted by the NRC, for each reactor vessel beltline material for the EOL fluence of the material.

- The assessment of RT_{PTS} must use the calculation procedures given in the PTS rule and must specify the bases for the projected value of RT_{PTS} for each beltline material. The report must specify the copper and nickel contents and the fluence values used in the calculation for each beltline material.

- This assessment must be updated whenever there is significant change in projected values of RT_{PTS}, or upon the request for a change in the expiration date for operation of the facility. Changes to RT_{PTS} values are significant if either the previous value or the current value, or both values, exceed the screening criterion prior to the expiration of the operating license, including any license renewal term, if applicable for the plant.

- The RT_{PTS} screening criteria values for the beltline region are:

  - 270°F for plates, forgings, and axial weld materials
  - 300°F for circumferential weld materials

- RT_{PTS} must be calculated for each vessel beltline material using a fluence value, f, which is the EOL fluence for the material.

Per 10 CFR Part 50.61, the following equations and variables are to be used for calculating EOL RT_{PTS} values at the clad/base metal interface of the vessel.

$$RT_{PTS} = RT_{NDT(U)} + M + \Delta RT_{PTS}$$

where,

$RT_{NDT(U)}$ = Initial RT_{NDT} value, °F

$$M = 2\sqrt{\sigma_i^2 + \sigma_\Delta^2} \text{ (°F)}$$

σ_i = 0°F when Initial RT_{NDT} is a measured value
σ_i = 17°F when Initial RT_{NDT} is a generic value

For plates and forgings:

σ_Δ = 17°F when surveillance capsule data is not used
σ_Δ = 8.5°F when credible surveillance capsule data is used

For welds:

σ_Δ = 28°F when surveillance capsule data is not used
σ_Δ = 14°F when credible surveillance capsule data is used

(σ_Δ not to exceed $0.5^*\Delta RT_{PTS}$)

$$\Delta RT_{PTS} = CF * f^{(0.28 - 0.10 \log f)}$$

where,

CF = chemistry factor (°F)
f = neutron fluence (10^{19} n/cm^2, E > 1.0 MeV) at the clad/base metal interface on the inside surface of the vessel

In accordance with 10 CFR Part 50.61, RT_{PTS} values under SPU conditions were calculated for the vessel beltline and materials at EOL (36 EFPY).

2.1.3.2.4 Results

Calculated RT_{PTS} values, and the interim calculations to obtain these values, are contained in Table 2.1.3-3. The limiting material is Lower Shell Plate R-1108-2 for Unit 1 and Intermediate Shell Plate R-3807-2 for Unit 2, with the more limiting RT_{PTS} value occurring for calculations using the Regulatory Guide 1.99, Revision 2, Position 1.1 Chemistry Factor, as opposed to the Position 2.1 Chemistry Factor calculated from surveillance data. The highest RT_{PTS} value at 36 EFPY is 99°F (for the Unit 1 Lower Shell Plate R-1108-2). This value is substantially below the NRC screening criteria for vessel plates of 270°F.

All of the beltline materials in the CPNPP Units 1 and 2 reactor vessels are below the RT_{PTS} screening criteria values of 270°F, for axially oriented welds, plates and forgings, and 300°F, for circumferentially oriented welds, at 36 EFPY.

Furthermore, since the reactor vessel inlet temperature at CPNPP Units 1 and 2 is being maintained between 525° and 590°F, the equations and results for predicting ΔRT_{NDT} and PTS reference temperature (RT_{PTS}) remain valid without any adjustments for temperature effects.

2.1.3.3 Conclusions

Luminant Power has reviewed the evaluation of the effects of the proposed SPU for CPNPP Units 1 and 2 and concludes that the evaluation has adequately addressed changes in neutron fluence and their effects on PTS. The RT_{PTS} value for the limiting material at EOL (36 EFPY) for both units is below the 10 CFR Part 50.61 screening criteria for plates, forgings, and axial welds. Luminant Power further concludes that the vessel integrity evaluation is appropriate to ensure that CPNPP Units 1 and 2 continue to meet the requirements of 10 CFR Part 50.61, and provides information to ensure continued compliance with GDC-14 and -31 in this respect following implementation of the proposed SPU. Therefore, Luminant Power finds the proposed SPU acceptable with respect to PTS.

2.1.3.4 References

1. WCAP-13422, "Analysis of Capsule U from the Texas Utilities Electric Company Comanche Peak Unit No. 1 Reactor Vessel Radiation Surveillance Program," July 1992.

2. WCAP-10684, "Texas Utilities Generating Company Comanche Peak Unit No. 2 Reactor Vessel Radiation Surveillance Program," October 1984.

Table 2.1.3-1

Maximum Calculated Neutron Exposure of the Reactor Vessel Beltline Materials at the Clad/Base Metal Interface (n/cm^2, (E > 1.0 MeV))

Projected EFPY	Azimuthal Location				
	0°	15°	21°	30°	45°
CPNPP Unit 1					
15.9[1]	6.26E+18	8.99E+18	9.90E+18	9.21E+18	9.13E+18
20.0	8.01E+18	1.14E+19	1.24E+19	1.13E+19	1.09E+19
25.0	1.02E+19	1.44E+19	1.55E+19	1.38E+19	1.31E+19
32.0	1.32E+19	1.86E+19	1.98E+19	1.74E+19	1.61E+19
36.0	1.49E+19	2.09E+19	2.23E+19	1.94E+19	1.78E+19
40.0	1.66E+19	2.33E+19	2.47E+19	2.14E+19	1.96E+19
48.0	2.00E+19	2.81E+19	2.97E+19	2.55E+19	2.30E+19
54.0	2.26E+19	3.17E+19	3.33E+19	2.85E+19	2.56E+19
CPNPP Unit 2					
14.5[2]	6.14E+18	8.51E+18	9.20E+18	8.37E+18	8.31E+18
20.0	8.51E+18	1.18E+19	1.26E+19	1.12E+19	1.07E+19
25.0	1.07E+19	1.48E+19	1.57E+19	1.37E+19	1.29E+19
32.0	1.37E+19	1.89E+19	2.00E+19	1.73E+19	1.59E+19
36.0	1.54E+19	2.13E+19	2.25E+19	1.93E+19	1.77E+19
40.0	1.71E+19	2.37E+19	2.49E+19	2.13E+19	1.94E+19
48.0	2.05E+19	2.85E+19	2.98E+19	2.54E+19	2.29E+19
54.0	2.31E+19	3.20E+19	3.35E+19	2.84E+19	2.55E+19

Notes:

1. Projected EOL 13, uprate to 3,612 MWt is assumed to occur at the onset of Cycle 14.
2. Projected EOL 11, uprate to 3,612 MWt is assumed to occur at the onset of Cycle 12.

Table 2.1.3-2 Summary of the CPNPP Units 1 and 2 Beltline Material Chemistry and Chemistry Factors Based on Regulatory Guide 1.99, Revision 2				
Beltline Materials	**Wt. % Cu**	**Wt. % Ni**	**Position 1.1 CF**	**Position 2.1 CF**
CPNPP Unit 1				
Intermediate Shell Plate R-1107-1	0.07	0.62	44.0°F	--
Intermediate Shell Plate R-1107-2	0.07	0.67	44.0°F	--
Intermediate Shell Plate R-1107-3	0.06	0.65	37.0°F	--
Lower Shell Plate R-1108-1	0.08	0.65	51.0°F	--
Lower Shell Plate R-1108-2	0.06	0.60	37.0°F	24.3°F
Lower Shell Plate R-1108-3	0.08	0.65	51.0°F	--
Beltline Region Weld Metal	0.045	0.20	46.0°F	15.7°F
CPNPP Unit 2				
Intermediate Shell Plate R-3807-1	0.06	0.64	37°F	--
Intermediate Shell Plate R-3807-2	0.06	0.64	37°F	21.6°F
Intermediate Shell Plate R-3807-3	0.05	0.60	31°F	--
Lower Shell Plate R-3816-1	0.05	0.59	31°F	--
Lower Shell Plate R-3816-2	0.03	0.65	20°F	--
Lower Shell Plate R-3816-3	0.04	0.63	26°F	--
Intermediate and Lower Shell Longitudinal Welds	0.046	0.059	31.5°F	32.8°F
Intermediate to Lower Shell Girth Weld	0.046	0.059	31.5°F	32.8°F

Table 2.1.3-3 RT_{PTS} Calculations for CPNPP Units 1 and 2 Beltline Materials at 36 EFPY								
Material	**RG 1.99 R2 *Method***	**CF (°F)**	**Fluence (10^{19} n/cm²)**	**FF(1)**	**ΔRT_{PTS}(2) (°F)**	**$RT_{NDT(U)}$(3) (°F)**	**Margin(4) (°F)**	**RT_{PTS}(5) (°F)**
CPNPP Unit 1								
Intermediate Shell Plate R-1107-1	Position 1.1	44.0	2.23	1.2173	53.56	10.0	34.0	98
Intermediate Shell Plate R-1107-2	Position 1.1	44.0	2.23	1.2173	53.56	-10.0	34.0	78
Intermediate Shell Plate R-1107-3	Position 1.1	37.0	2.23	1.2173	45.04	10.0	34.0	89
Lower Shell Plate R-1108-1	Position 1.1	51.0	2.23	1.2173	62.08	0.0	34.0	96
Lower Shell Plate R-1108-2	Position 1.1	37.0	2.23	1.2173	45.04	20.0	34.0	99
	Position 2.1	24.3	2.23	1.2173	29.58	20.0	29.58	79
Lower Shell Plate R-1108-3	Position 1.1	51.0	2.23	1.2173	62.08	0.0	34.0	96
Beltline Region Weld Metal	Position 1.1	46.0	2.23	1.2173	56.00	-70.0	56.00	42
	Position 2.1	15.7	2.23	1.2173	19.11	-70.0	19.11	-32
CPNPP Unit 2								
Intermediate Shell Plate R-3807-1	Position 1.1	37.0	2.25	1.2196	45.12	-20	34.00	59
Intermediate Shell Plate R-3807-2	Position 1.1	37.0	2.25	1.2196	45.12	10	34.00	89
	Position 2.1	21.6	2.25	1.2196	26.34	10	26.34	63
Intermediate Shell Plate R-3807-3	Position 1.1	31.0	2.25	1.2196	37.81	-20	34.00	52
Lower Shell Plate R-3816-1	Position 1.1	31.0	2.25	1.2196	37.81	-30	34.00	42
Lower Shell Plate R-3816-2	Position 1.1	20.0	2.25	1.2196	24.39	0	24.39	49
Lower Shell Plate R-3816-3	Position 1.1	26.0	2.25	1.2196	31.71	-40	31.71	23

Material	RG 1.99 R2 Method	CF (°F)	Fluence (10^{19} n/cm^2)	FF(1)	ΔRT_{PTS}(2) (°F)	$RT_{NDT(U)}$(3) (°F)	Margin(4) (°F)	RT_{PTS}(5) (°F)
CPNPP Unit 2 (cont.)								
Intermediate and Lower Shell Longitudinal Welds	Position 1.1	31.5	2.25	1.2196	38.42	-50	38.42	27
	Position 2.1	32.8	2.25	1.2196	40.00	-50	28.00	18
Intermediate to Lower Shell Girth Weld	Position 1.1	31.5	2.25	1.2196	38.42	-60	38.42	17
	Position 2.1	32.8	2.25	1.2196	40.00	-60	28.00	8

Table 2.1.3-3 (cont.)

RT_{PTS} Calculations for CPNPP Units 1 and 2 Beltline Materials at 36 EFPY

Notes:

1. FF = fluence factor = $f^{(0.28 - 0.1 \log (f))}$
2. ΔRT_{PTS} = CF * FF
3. Initial RT_{NDT} values are measured values
4. $M = 2 *(\sigma_i^2 + \sigma_\Delta^2)^{1/2}$
5. $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS}$ + Margin (°F)

2.1.4 Reactor Internals and Core Support Materials

2.1.4.1 Regulatory Evaluation

The reactor internals and core supports include structures, systems, and components (SSCs) that perform safety functions or whose failure could affect safety functions performed by other SSCs. These safety functions include:

- Reactivity monitoring and control
- Core cooling
- Fission product confinement (within both the fuel cladding and the reactor coolant system (RCS))

The Luminant Power review covered:

- The materials' specifications and mechanical properties
- Welds
- Weld controls
- Non-destructive examination (NDE) procedures
- Corrosion resistance
- Susceptibility to degradation

The acceptance criteria are based on:

- General Design Criterion (GDC)-1 and 10 CFR 50 Part 55a for material specifications, controls on welding, and inspection of reactor internals and core supports.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the adequacy of design relative to:

- GDC-1 is described in the FSAR Section 3.1.1.1, General Design Criterion 1 - Quality Standards and Records.

  The systems and components of the facility are classified according to their importance in the prevention and mitigation of accidents. Reactor components use the classification system developed by American National Standards Institute (ANSI) N18.2-1973, Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants. Classifications and any deviations are described in FSAR Section 3.2. Each component is given a safety class designation.

FSAR Chapter 17 provides direct reference to the Quality Assurance Program established to provide assurance that safety-related SSCs satisfactorily perform their intended safety functions. The procedures for generating and maintaining appropriate design, fabrication, erection, and testing records are contained within the referenced documents.

Quality standards applicable to safety related SSCs are generally contained in codes such as the American Society of Mechanical Engineers Boiler and Pressure Vessel (ASME B&PV) Code. The applicability of these codes is specifically identified throughout this report and is summarized in FSAR Section 3.2.2. The procedures for generating and maintaining appropriate design, fabrication, erection, and testing records are contained within the referenced documents.

The specifications for materials used for reactor vessel internal (RVI) components are shown in FSAR Table 5.2-3. FSAR Section 4.5.2 provides information on RVI materials, controls on welding, and the cleaning and fabrication of stainless steel RVI components. FSAR Section 5.2.3.4.4 summarizes the 4-point program designed to prevent intergranular attack of austenitic stainless steel components.

- 10 CFR Part 50.55(a) is described in FSAR Section 5.2.1.1, Compliance with 10 CFR Part 50.55(a). RCS components are designed and fabricated in accordance with 10 CFR Part 50.55a. The actual addenda of the ASME Code applied in the original design of each component are listed in FSAR Table 5.2-1.

Details of the RVI and their design conditions are provided in FSAR Sections 3.9N.1, 3.9N.5, and 4.5.

Section 3.9N.5.1 of the FSAR states:

- The components of the reactor internals are divided into three parts, consisting of the lower core support assembly, (including the entire core barrel and neutron shield pad assembly), the upper core support assembly and the incore instrumentation support structure. The reactor internals support the core, maintain fuel alignment, limit fuel assembly movement, maintain alignment between fuel assemblies and CRDM's, direct coolant flow past the fuel elements, direct coolant flow to the pressure vessel head, provide gamma and neutron shielding, and provide guides for the in-core instrumentation.

- The major containment and support member of the reactor internals is the lower core support assembly, shown in Figure 3.9N-9. This assembly consists of the core barrel, the core baffle, the lower core plate and support columns, the neutron shield pads, and the core support, which is welded to the core barrel. The major material for this assembly is Type 304 stainless steel. The lower core support assembly is supported at its upper flange from a ledge in the reactor vessel flange and its lower end is restrained in its transverse movement by a radial support system attached to the vessel wall.

The neutron shield pad design consists of four sets of stainless steel plates strategically placed on the core barrel in areas of peak fast neutron flux on the reactor pressure vessel. See Figure 3.9N-9. Attachment of each of the plate sections to the core barrel is accomplished through a series of sixteen 7/8-inch stainless steel bolts and three 2-3/8-inch stainless steel pins.

- The upper core support assembly, shown in Figure 3.9N-10 and 3.9N-11 consists of the upper support, the upper core support plate, the support columns, and the guide tube assemblies. The support columns establish the spacing between the upper support and the upper core plate. They are fastened at the top and bottom to these plates. The support columns transmit the mechanical loading between the two plates and serve the supplementary function of supporting thermocouples.

2.1.4.2 Technical Evaluation

2.1.4.2.1 Introduction

This section of the report summarizes the evaluations, and their results, of the potential materials degradation issues arising from the effect of stretch power uprate (SPU) on the performance of reactor internals and core support materials at the Comanche Peak Nuclear Power Plant (CPNPP).

The Westinghouse Owners Group (WOG) Life Cycle Management and License Renewal Program prepared topical report WCAP–14577 Rev. 1-A, License Renewal Evaluation: Aging Management for Reactor Internals, (Reference 1). The topical report describes the aging degradation mechanisms to determine the aging effects. All identified effects are evaluated to determine that the aging effects are being managed to ensure RVI components perform their intended functions. The evaluation also included the time-limited aging analyses (TLAAs).

The NRC review of the WOG topical report concluded that the report provides an acceptable demonstration that the applicable effects of aging on reactor vessel internals components will be adequately managed for the WOG plants, such that there is a reasonable assurance that the RVI components will perform their intended functions in accordance with the current licensing basis during the remainder of the base licensing period. The SPU evaluation considered potential changes in the aging effects due to the change in the service conditions resulting from the proposed SPU conditions. These are considered below.

The primary objective of the SPU assessment was to ensure that the new SPU environmental conditions (chemistry, temperature, and fluence) do not introduce any new aging effects on the RVI components during the remainder of the base licensing period, nor change the manner in which the component aging is managed by the aging management program credited in the topical report WCAP-14577, Rev. 1-A (Reference 1), and accepted by the NRC in the Safety Evaluation Report (SER).

The relevant potentially impacted degradation (aging) mechanisms are:

- Integrity of reactor vessel fuel cladding materials
- Transgranular stress corrosion cracking (TGSCC) and intergranular stress corrosion cracking (IGSCC) of stainless steels
- Primary water stress corrosion cracking (PWSCC) of Alloy 600 and Alloy X-750 components
- Neutron irradiation embrittlement and void swelling of austenitic steel material internals
- Irradiation-assisted stress corrosion cracking (IASCC) of stainless steels

An assessment of these aging mechanisms is considered in the following subsections.

2.1.4.2.2 Input Parameters, Assumptions, and Acceptance Criteria

Proposed SPU Service Conditions

The SPU will cause the following changes in the RCS chemistry conditions (Reference 2), neutron fluence levels, and temperatures provided in Licensing Report (LR) Section 1.1:

- The reactor coolant lithium/boron chemistry program is coordinated such that a target pH range between 7.19 and 7.40 is maintained with an initial target lithium level of 5.71 ppm. The lithium level is then decreased gradually during the fuel cycle as the boron diminishes, thus maintaining a target pH value of 7.40 through the end of the fuel cycle (Westinghouse chemistry guidelines recommends the at temperature pH to be maintained between 6.90 and 7.40).
- For Unit 1, the estimated maximum fast neutron (E>0.1 MeV) exposure of the reactor internals for operating periods of 36 effective full-power years (EFPYs) are summarized in Table 2.1.4-1. The values shown for 15.9 EFPY and the SPU have been extrapolated from calculations of the reactor pressure vessel fluence and were based, in part, on work that was completed to support pressure vessel integrity evaluations for the SPU. These maximum exposures occur on the inside surface of the baffle plates opposite the central sections of the reactor core. The estimated exposures as a result of the SPU are compared to the design basis values as well as the estimated exposures at the end of Cycle 13 (15.9 EFPY). Note that the estimated exposures as a result of the SPU are less than the design basis values for 36 EFPY. This occurs due to the use of low leakage cores.
- For Unit 1, a maximum increase in the peak steady-state service temperature of 1.2°F at the reactor vessel hot leg location and a decrease in service temperature of 1.2°F at the reactor vessel cold leg and bottom-mounted instrumentation (BMI) penetration locations will occur due to the SPU. This is summarized for Unit 1 in Table 2.1.4-2.

- For Unit 2, the estimated maximum fast neutron (E>0.1 MeV) exposure of the reactor internals for operating periods of 36 EFPY are summarized in Table 2.1.4-3. The values shown for 14.5 EFPY and the SPU have been extrapolated from calculations of the reactor pressure vessel fluence and were based, in part, on work that was completed to support pressure vessel integrity evaluations for the SPU. These maximum exposures occur on the inside surface of the baffle plates opposite the central sections of the reactor core. The estimated exposures as a result of the SPU are compared to the design basis values as well as the estimated exposures at the end of Cycle 11 (14.5 EFPY). Note that the estimated exposures as a result of the SPU are less than the design basis values for 36 EFPY. This occurs due to the use of low leakage cores.

- For Unit 2, a maximum increase ΔT in the peak steady-state service temperature of 1.8°F at the reactor vessel hot leg location and a decrease ΔT in service temperature of 1.8°F at the reactor vessel cold leg and BMI penetration locations will occur due to the SPU. This is summarized for Unit 2 in Table 2.1.4-4.

2.1.4.2.3 Description of Analyses and Evaluations

The effect of changes in service conditions due to the proposed SPU on the performance of the reactor vessel internals materials is discussed in the following paragraphs.

Materials Specifications, Weld Controls, and NDE Inspections

The NRC's acceptance criteria for reactor internals and core support materials are based on GDC-1 and 10 CFR 50.55a for material specifications, controls on welding, and inspection of reactor internals and core supports. The review of CPNPP covered the materials' specifications and mechanical properties, welds, weld controls, NDE procedures, corrosion resistance, and susceptibility to degradation. Specific review criteria are contained in WCAP-14577, Rev. 1-A. The proposed SPU is not expected to cause negative effects.

Stress Corrosion Cracking

The two degradation mechanisms that are operative in the internals austenitic stainless steels are IGSCC and TGSCC. Susceptible materials, sensitized microstructure, tensile stress, and the presence of oxygen are required for the occurrence of IGSCC, while the introduction of halogens such as chlorides and the presence of oxygen are prerequisites for the occurrence of TGSCC. The principal method of preventing IGSCC and TGSCC is by water chemistry control. The reactor coolant chemistry must be rigorously controlled, particularly with regard to oxygen, chlorides, and other halogens. Ingress from other species, such as demineralizer resins, is carefully monitored, and corrective actions are taken to preclude exposure. The minimal increase in temperature due to the SPU would not affect IGSCC or TGSCC of austenitic stainless steels.

PWSCC is another form of IGSCC degradation that has been observed in Alloy 600 and Alloy X-750 materials in pressurized water reactor (PWR) applications. The rod cluster control

assembly (RCCA) guide tube support pins and clevis insert bolts are fabricated from X-750 material. The clevis inserts are manufactured from Alloy 600 material.

The cracking of X-750 material is attributed to a combination of high stress and undesirable microstructure. The heat treatment specification for the replacement split pin material and the support pin design at CPNPP was to provide a more PWSCC resistant microstructure and lower stress condition. The Alloy X-750 clevis insert bolts in older plant designs experienced cracking in some plants after approximately 13 years of operation. However, the degradation of clevis insert bolts would not result in a loss of intended function since the design geometry is such that the insert sits in a constrained groove and degradation of the bolts would not cause the displacement of the clevis insert from its original position. The slight decrease in temperature due to the SPU would decrease the potential degradation of the X-750 material.

The Alloy 600 clevis inserts experience lower fluence, temperature, and stresses in comparison to the support pins. The clevis inserts experience essentially compressive stress and no failures have been reported. Furthermore, like the clevis insert bolts, a failure of the clevis inserts would not result in a loss of intended function due to the nature of the design. Therefore, the effects of PWSCC on the clevis inserts are not significant. The slight temperature decrease would ultimately be beneficial to the Alloy 600 clevis inserts.

WCAP-14577, Rev. 1-A (Reference 1), considered the potential stress corrosion cracking (SCC) degradation and concluded that the effects of all forms of SCC are not significant for Alloy 600, X-750, and stainless steel RVI components. The NRC review of the topical report concluded that there is a reasonable assurance that the RVI components will perform their intended functions in accordance with the current licensing basis.

The chemistry program at CPNPP operates at an elevated pH level, while the lithium level has an initial target value of 5.71 ppm (Reference 2). The chemistry program for the SPU does not involve an introduction of any of the contributors (stress, oxygen, or halogen) for SCC cracking, therefore no impact on the SCC material degradation is expected in the RVI components as a result of the SPU.

Fuel-Cladding Corrosion Effects

The CPNPP SPU lithium, boron, and pH management program was reviewed. The chemistry program operates at an elevated pH level, while the lithium level has an initial target value of 5.71 ppm. Experience with operating plants as well as with the guidelines provided by EPRI (Reference 3) suggests that increasing initial lithium concentrations up to 3.5 ppm with controlled boron concentrations to maintain pH values ranging from 6.9 to 7.4 has not produced any undesirable material integrity issues that could be statistically defined from the database of laboratory results available in 2003.

CPNPP Units 1 and 2 are two of the lead units in this ongoing EPRI/Industry study of the impact of operating plants at high lithium levels (References 4, 5, and 6).

Irradiation Embrittlement

Irradiation embrittlement is possible in the reactor internals components fabricated from austenitic stainless steel and nickel-based alloys with expected neutron fluences in excess of 1×10^{21} n/cm^2 (E > 0.1 MeV). If the expected neutron fluence is less than approximately 1×10^{21} n/cm^2 (E > 0.1 MeV), then the changes in mechanical properties due to neutron exposure are insignificant. The reactor internals components with fluences greater than 1×10^{21} n/cm^2 (E > 0.1 MeV) (such as lower core barrel, baffle/former assembly, baffle/former bolts, lower core plate and fuel pins, lower support forging, and clevis bolts) are potentially susceptible to irradiation embrittlement.

The SPU expected maximum fast neutron exposure levels of the reactor internals for operating periods of 36 EFPY are listed in LR subsection 2.1.4.2.2 above. Experience has shown that the following RVI components are exposed to the highest in-core neutron radiation fields and hence are most susceptible to crack initiation and growth due to IASCC and loss of fracture toughness due to neutron irradiation embrittlement and/or void swelling:

- Lower core plate and fuel alignment pins
- Lower support columns
- Core barrel and core barrel flange in active core region
- Thermal shield
- Bolting-lower support column, baffle-former, and barrel-former

Data from power reactor irradiation of Type 304 and Type 316 stainless steel are available from several studies (References 7, 8, and 9). Embrittlement, as evidenced by increases in yield strength and decreases in uniform and total elongation, is common in these materials after irradiation. Studies (References 7 and 8) showed that embrittlement of stainless steel can occur at fluences as low as 1×10^{21} n/cm^2 (E > 0.1 MeV) in the more susceptible stainless steel materials such as 304SS. These same studies showed that the rate of change in mechanical properties is reduced at fluences above 2×10^{22} n/cm^2 (E > 0.1 MeV).

No instance of service related internals degradation has been recorded that can be directly attributed to irradiation embrittlement. However, the end-of-life fluence level for some internals components at CPNPP is approximately 1×10^{23} n/cm^2 (E > 0.1 MeV), therefore Luminant Power has committed to follow the industry Materials Reliability Program/Issues Task Group efforts on reactor internals and monitor developments in this area.

There are a number of industry activities currently underway to characterize and address aging effects on reactor vessel internals under the EPRI Materials Reliability Project (MRP). As a result of these efforts, further understanding of these aging effects is being developed by the industry to provide additional bases for whether inspections over and beyond those currently required by ASME Section XI should be implemented. The MRP strategy is to evaluate potential aging mechanisms and their effects on specific RVI parts by evaluating causal parameters such as fluence, material properties, state of stress, and so forth. Critical locations can thereby be identified and tailored inspections can be conducted on either an integrated industry, nuclear steam supply system (NSSS), or plant-specific basis. The MRP projects

include material testing of baffle/former bolts removed from the Point Beach, Farley, and Ginna nuclear power plants and determination of bolt operating parameters.

Void Swelling

Void swelling is the gradual increase in size (physical dimension) of the RVI stainless steel component caused by the formation and growth of helium-vacancy clusters into voids due to the effect of irradiation. Although the effects of swelling can be potentially significant for those components which experience significant neutron irradiation while operating at elevated temperatures, the actual plant operations do not appear to produce the conditions necessary for significant swelling. At CPNPP, the SPU results in a minimal increase in the internals temperature therefore the necessary conditions for significant swelling do not increase. Recent data from Point Beach and Farley suggested very small (0.01 to 0.03 percent) amounts of swelling in baffle bolts. Extrapolation of these data suggests no concern with respect to void swelling until the end of extended life in U.S. PWRs. Fuel management schemes to reduce neutron leakage from the core have reduced one of the major factors contributing to swelling, and mechanisms such as creep and stress relaxation serve to reduce some of the adverse effects. WCAP-14577 (Reference 1) examined the effects of swelling and concluded that any actual swelling of the susceptible internals will not prevent them from performing their intended function during the license renewal period.

Industry data on swelling are currently being evaluated as part of the WOG and MRP. At present there have been no indications from the different bolt removal programs or functional "evaluations" that there are any discernible effects attributable to swelling. Luminant Power continues to participate and follow up industry efforts to investigate swelling effects on the reactor vessel internals.

Thermal Aging

Thermal aging of cast austenitic stainless steel can lead to precipitation of additional phases in the ferrite and growth of existing carbides at the ferrite/austenitic boundaries that can result in loss of ductility and fracture toughness of the material. The susceptibility to thermal aging is a function of the material chemistry, aging temperature, and time at temperature. All the cast duplex stainless steel reactor internals in the Westinghouse-designed NSSS are made from CF-8 or CF-8A materials which contain low or zero molybdenum and are less susceptible to thermal aging than the molybdenum-containing grades.

The Unit 1 reactor internals contain some cast austenitic stainless steel material. Although this material is potentially susceptible to thermal aging embrittlement under prolonged exposure to elevated temperature, the chemistry content and the service temperature at Unit 1 are not favorable to produce enough loss of toughness to have any significant impact on the structural integrity.

WCAP-14577 (Reference 1) conducted an evaluation of the effects of thermal aging and concluded that the effects of thermal aging are insignificant to all of the reactor internals

components and aging management of this effect is not required even through an extended period of operation.

The Unit 2 reactor internals do not contain cast austenitic stainless steel material.

2.1.4.2.4 Results

The results of the potential material degradation assessment of the RVIs showed that no materials degradation issues result from the proposed SPU at CPNPP. On this basis, it is concluded that the new SPU environmental conditions (temperature and fluence) do not introduce any new aging effects on their components during the current licensed operation, nor does the SPU change the manner in which the component aging is managed by the aging management program credited in WCAP-14577, Rev. 1-A.

2.1.4.3 Conclusions

Luminant Power has reviewed the evaluation of the effects of the proposed SPU on the susceptibility of reactor internal and core support materials to known degradation mechanisms and concludes that the evaluation has identified appropriate aging management programs to address the effects of changes in operating temperature and neutron fluence on the integrity of reactor internal and core support materials. Luminant Power further concludes that the evaluation has demonstrated that the reactor internal and core support materials will continue to be acceptable and will continue to meet the requirements of GDC-1 and 10 CFR 50.55a following implementation of the proposed SPU. Therefore, Luminant Power finds the proposed SPU acceptable with respect to reactor internal and core support materials.

2.1.4.4 References

1. WCAP-14577, Rev. 1-A, “License Renewal Evaluation: Aging Management for Reactor Internals,” March 2001.

2 TXX-02037, Supplement to License Amendment Request for Lead Test Assemblies February 18, 2002.

3. EPRI TR-1002884, Volume 1, “Pressurized Water Reactor Primary Water Chemistry Guidelines,” Rev. 5, September 2003.

4. Assessment of Elevated pH Program for Comanche Peak Steam Electric Station, EPRI, Palo Alto, CA, 2001, TR-1006282.

5. WCAP-16247, Evaluation of the Impact of High pH on Primary System Materials Corrosion, Final Report for WOG Materials Subcommittee Program MUHP-5075, March 2004.

6. J. Stevens, D. Farnsworth, J. Bosma and J. Deshon, “Elevated RCS pH Program at Comanche Peak,” Int'l Conf. Water Chem. Nucl. Reactor Syst., Jeju Island, Korea, October 23 – 26, 2006, Paper P 1.1

7. Kangilaski, M., “The Effects of Neutron Radiation on Structural Materials,” REIC Report No. 45, Radiation Effects Information Center, Battelle Memorial Institute, Columbus, Ohio (June 1967).

8. Robbins, R. E., et. al., “Post Irradiation Tensile Properties of Annealed and Cold Worked AISI – 304 Stainless Steel,” Trans American Nuclear Society, pp 488-489 (Nov. 1967).

9. Bloom, E. E., “Mechanical Properties of Materials in Fusion Reactor First-Wall and Blanket Systems,” Journal of Nuclear Materials, 85 and 86, pp 795-804 (1979).

Table 2.1.4-1 Unit 1 Summary of Fluence Changes in the RVIs Due to the Proposed SPU			
Operating Time (EFPY)	**Fluence (E > 0.1 MeV) (n/cm^2) (Current Design Basis)**	**Fluence (E > 0.1 MeV) (n/cm^2) (Current, 15.9 EFPY, EOC 13)**	**Fluence (E > 0.1 MeV) (n/cm^2) (Uprate)**
36	1.8E+23	5.10E4.57E+22	1.09E08E+23

Table 2.1.4-2 Unit 1 Summary of Service Temperature Changes in the RV Hot and Cold Legs Due to the Proposed SPU			
Core Power Level (MWt)	**Penetration Location**	**Temperature (°F)**	**Maximum Change in the Steady-State Peak Temperature Due to Uprating (°F)**
3,458 (Current)	RV Hot Leg	619.2	
3,458 (Current)	RV Cold Leg, RVH, and BMI	559.2	
3,612 (SPU)	RV Hot Leg	606.2 – 620.4	1.2
3,612 (SPU)	RV Cold Leg, RVH, and BMI	542.2 – 558.0	-1.2

Table 2.1.4-3 Unit 2 Summary of Fluence Changes in the RVIs Due to the Proposed SPU			
Operating Time (EFPY)	**Fluence (E > 0.1 MeV) (n/cm^2) (Current Design Basis)**	**Fluence (E > 0.1 MeV) (n/cm^2) (Current, 14.5 EFPY, EOC 11)**	**Fluence (E > 0.1 MeV) (n/cm^2) (4.4% Uprate)**
36	1.8E+23	5.10E4.57E+22	1.09E08E+23

Table 2.1.4-4 Unit 2 Summary of Service Temperature Changes in the RV Hot and Cold Legs Due to the Proposed SPU			
Core Power Level (MWt)	**Penetration Location**	**Temperature (°F)**	**Maximum Change in the Steady-State Peak Temperature Due to Uprating (°F)**
3,458 (Current)	RV Hot Leg	618.6	
3,458 (Current)	RV Cold Leg, RVH, and BMI	559.8	
3,612 (SPU)	RV Hot Leg	606.2 – 620.4	1.8
3,612 (SPU)	RV Cold Leg, RVH, and BMI	542.2 – 558.0	-1.8

2.1.5 Reactor Coolant Pressure Boundary Materials

2.1.5.1 Regulatory Evaluation

The reactor coolant pressure boundary (RCPB) defines the boundary of systems and components containing the high pressure fluids produced in the reactor. The Luminant Power review of RCPB materials covered their specification, compatibility with the reactor coolant, fabrication and processing, susceptibility to degradation, and degradation management programs.

The acceptance criteria for this review are:

- 10 CFR 50 Part 55a and General Design Criterion (GDC)-1, insofar as they require that structures, systems, and components (SSCs) important to safety be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed.
- GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.
- GDC-14, insofar as it requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating fracture.
- GDC-31, insofar as it requires that the RCPB be designed with margin sufficient to assure that, under specific conditions, it will behave in a non-brittle manner and the probability of a rapidly propagating fracture is minimized.
- 10 CFR Part 50, Appendix G, which specifies fracture toughness requirements for ferritic components of the RCPB.

Additional guidance for primary water stress corrosion cracking (PWSCC) of dissimilar metal welds and associated inspection programs is contained in Nuclear Regulatory Commission (NRC) Generic Letter (GL) 97-01 and NRC Bulletins (BL) 01-01, BL-02-01, and BL-02-02. Additional review guidance for thermal embrittlement of cast austenitic stainless steel components is contained in a letter from C. Grimes, NRC, to D. Walters, Nuclear Energy Institute (NEI), dated 19 May, 2000.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the adequacy of pressure-retaining components and component supports' design relative to:

- 10 CFR Part 50.55a is described in FSAR Section 5.2.1.1, Compliance with 10 CFR Part 50.55a. Reactor coolant system (RCS) components are designed and fabricated in accordance with 10 CFR Part 50.55a. The actual addenda of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME B&PV) Code applied in the original design of each component are listed in FSAR Table 5.2-1.

- GDC-1, Quality Standards and Records, is described in FSAR Section 3.1.1.1.

  The SSCs of the facility are classified according to their importance in the prevention and mitigation of accidents. Reactor components use the classification system developed by American National Standards Institute (ANSI) N18.2-1973, Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants. Classifications and any deviations are described in FSAR Section 3.2. Each component is given a safety class designation.

  FSAR Chapter 17 provides direct reference to the Quality Assurance (QA) Program established to provide assurance that safety-related SSCs satisfactorily perform their intended safety functions. The procedures for generating and maintaining appropriate design, fabrication, erection, and testing records are contained within the referenced documents.

- GDC-4 is described in the FSAR Section 3.1.1.4, Environmental and Dynamic Effect Bases.

  The station's SSCs important to safety are designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents (LOCAs). Environmental conditions are described in FSAR Section 3.11.

  These SSCs are appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that can result from equipment failures and from events and conditions outside the nuclear power unit.

  Details of the design, environmental testing, and construction of these SSCs are included in FSAR Chapters 3, 5, 6, 7, 8, 9, and 10. Evaluation of the performance of safety features is contained in FSAR Chapter 15.

  The leak-before-break (LBB) methodology demonstrates that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping. It has been applied to CPNPP Units 1 and 2 to exclude from the design basis the dynamic effects of postulated pipe ruptures in the primary coolant loop piping and 10-inch and larger reactor coolant loop branch lines, as discussed in FSAR Sections 3.6B.2.5.1 and 3.6B.2.5. Implementation of this technology eliminates the

need for pipe whip restraints and jet impingement barriers, respectively. Containment design, emergency core cooling, and environmental qualification requirements are not influenced by this modification.

- GDC-14, Reactor Coolant Pressure Boundary, is described in FSAR Section 3.1.2.5.

  The RCS pressure boundary is designed to accommodate the system pressures and temperatures attained under the expected modes of plant operation, including anticipated transients, and to maintain the stresses within applicable stress limits (FSAR Section 5.2). Also, RCPB materials and selection and fabrication techniques ensure a low probability of gross rupture of significant leakage.

  In addition to the loads imposed on the system under normal operating conditions, consideration is also given to abnormal loading conditions, such as seismic and pipe rupture, which are discussed in FSAR Sections 3.6 and 3.7.

  The LBB methodology demonstrates that the probability of rupturing primary coolant piping is extremely low under design basis conditions. LBB methodology has been applied to CPNPP Units 1 and 2 to exclude from the design basis the dynamic effects of postulated ruptures in the primary coolant loop piping, as discussed in FSAR Section 3.6B.2.5.1. Implementation of this technology eliminates the need for primary coolant loop piping whip restraints and jet impingement barriers. Containment design, emergency core cooling, and environmental qualification requirements are not influenced by the application of LBB methodology.

- GDC 31, Fracture Prevention of Reactor Coolant Pressure Boundary, is described in FSAR Section 3.1.4.2.

  Close control is maintained over material selection and fabrication for the RCS to assure that the boundary behaves in a non-brittle manner. The RCS materials exposed to the coolant are corrosion-resistant stainless steel or Inconel. The reference temperature (RT_{NDT}) of the RV structural steel is established by Charpy V-notch and drop weight tests, in accordance with 10 CFR Part 50, Appendix G.

  The fabrication and quality control techniques used in the fabrication of the RCS are consistent with those used for the RV. The inspection of RV, pressurizer, piping, pumps, and steam generator are governed by ASME Code requirements.

The RCS is described in FSAR Section 5.1. The RCS and the RCPB are shown in FSAR Figure 5.1-1. RCPB components include the following equipment, which is designed to the ASME B&PV Code, Section III, Class 1 requirements:

- Reactor vessel, including control rod drive mechanism (CRDM) housings

- Steam generators (reactor coolant side)

- Pressurizer (and surge line attached to one of the reactor coolant loops)
- Reactor coolant pumps (RCPs)
- Pressurizer relief tank (PRT)
- Safety and relief valves
- Reactor coolant piping
- Interconnecting piping, valves and fittings between the principal components described above
- The piping, fittings, and valves leading to connecting auxiliary or support systems

The RCS consists of four similar heat transfer loops connected in parallel to the reactor pressure vessel (RPV). Each loop contains a RCP and a steam generator. In addition, the system includes a pressurizer, a PRT, interconnecting piping, valves, and instruments necessary for operational control.

The RCPB materials are addressed in Section 5.2.3 and Table 5.2-3 of the FSAR. The RCPB materials were selected for the expected environmental and service conditions. They have been designed, procured, fabricated, and inspected to satisfy the requirements of ASME Section VIII, Division I.

2.1.5.2 Technical Evaluation

2.1.5.2.1 Introduction

This section of the report summarizes the evaluations and results of the potential materials degradation issues arising from the effect of the CPNPP stretch power uprate (SPU) on the performance of RCPB component materials.

The SPU evaluation assessed the potential effect of changes in the RCS chemistry (impurities), pH conditions, and SPU service temperatures on the integrity of primary component pressure boundary materials during service. The evaluation includes:

- An assessment of the potential effect of water chemistry changes on the general corrosion (wastage) of carbon steel components, stress corrosion cracking (SCC) of system austenitic stainless steel materials, and the management strategy of any associated issues.
- An assessment of the effect of change in the service temperature on PWSCC of Alloy 600/182/82 nickel base alloys, thermal aging of cast austenitic stainless steel (CASS) materials, and the management strategy of any associated issues.

These assessments are discussed in the following subsections.

2.1.5.2.2 Input Parameters, Assumptions, and Acceptance Criteria

Proposed SPU Service Conditions

A review of the SPU design parameters indicates that the following changes in the RCS chemistry and service temperature conditions occur during operations after the SPU implementation:

- The reactor coolant lithium/boron chemistry program is coordinated such that a target pH range between 7.19 and 7.40 is maintained with an initial target lithium level of 5.71 ppm. The lithium level is then decreased gradually during the fuel cycle as the boron diminishes, thus maintaining a target pH value of 7.40 through the end of the fuel cycle. This is the current chemistry program employed at CPNPP.
- For Unit 1, a maximum increase in the peak steady-state service temperature of 1.2°F at the reactor vessel hot leg location and a decrease in service temperature of 1.2°F at the reactor vessel cold leg and bottom-mounted instrumentation (BMI) penetration locations will occur due to the SPU. This is summarized in Table 2.1.5-1.
- For Unit 2, a maximum increase in the peak steady-state service temperature of 1.8°F at the reactor vessel hot leg location and a decrease in service temperature of 1.8°F at the reactor vessel cold leg and BMI penetration locations will occur due to the SPU. This is summarized in Table 2.1.5-2.

2.1.5.2.3 Description of Analyses and Evaluations

The effect of change in service conditions (temperature and water chemistry) due to the proposed SPU on the performance of the RCPB materials is discussed in the following paragraphs.

General Corrosion/Wastage of Carbon Steel Components

The SPU reactor coolant lithium/boron program is coordinated such that an elevated pH value is maintained during the fuel cycle while maintaining a lithium level target value of 5.71 ppm.

The CPNPP Boric Acid Corrosion Control (BACC) Program is discussed in Section B2 1.6 of WCAP-14575, Licensing Renewal Evaluation: Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components (Reference 1). Inspections are routinely performed to verify the integrity of the reactor vessel head incorporating lessons learned from Davis-Besse, and to address NRC generic communications.

SCC of Austenitic Stainless Steels

The two degradation mechanisms that are operative in the pressure boundary austenitic stainless steel (base and weld) materials in the RCPB are intergranular SCC (IGSCC) and transgranular SCC (TGSCC). Susceptible materials, sensitized microstructure, and the presence of oxygen are required for the occurrence of IGSCC, while the introduction of halogens such as chlorides and the presence of oxygen are prerequisites for the occurrence of TGSCC.

The SPU reactor coolant lithium/boron program is coordinated such that an elevated pH value is maintained during the fuel cycle with an initial lithium level target value of 5.71 ppm.

The chemistry changes resulting from the SPU do not involve introduction of any of these contributors so that no effect on material degradation is expected in the stainless steel components as a result of the SPU. There is a negligible increase in material degradation due to the increased temperature change.

<u>Alloy 600/82/182 Components at Unit 1</u>

- Alloy 600 and Alloy 82/182 weld deposits are present in the CPNPP Unit 1 RCS at the following locations:
  - Reactor vessel inlet nozzle 82/182 welds
  - Reactor vessel outlet nozzle 82/182 welds
  - Pressurizer surge, spray, safety, and relief nozzle 82/182 welds
  - Alloy 600 BMI nozzles and Alloy 82/182 J-groove welds
  - Reactor vessel flange leakage monitor tube
  - Alloy 600 reactor vessel core support blocks and 82/182 welds
  - Alloy 82/182 reactor vessel core guide lug shell cladding
  - Alloy 600 reactor internals clevis inserts
- Alloy 690 and Alloy 52/152 weld deposits are present at the following locations:
  - CRDM nozzles, vent nozzle, and instrument nozzles in the replacement reactor vessel head. All nozzles are Alloy 690 TT (thermally treated) and welded to the inside diameter (ID) of the head with partial penetration welds using Alloy 52 weld deposits.
  - U-tubes of the Unit 1 steam generators (SGs). The tubing for the Unit 1 SGs is Alloy 690 TT.
  - Steam generator tubesheet cladding
  - Steam generator partition plate
  - Steam generator tube-to-tubesheet welds (autogenous)

Alloy 600/82/182 Components at Unit 2

- Alloy 600 and Alloy 82/182 weld deposits are present in the Unit 2 RCS at the following locations:
  - Reactor vessel upper head CRDMs and head vent penetrations. The penetrations are Alloy 600, welded to the ID of the head with partial penetration welds using 82/182 weld deposits
  - Alloy 600 BMI nozzles and Alloy 82/182 J-groove welds
  - Reactor vessel flange leakage monitor tube
  - Reactor vessel outlet nozzle 82/182 welds
  - Reactor vessel inlet nozzle 82/182 welds
  - Alloy 600 reactor vessel core support blocks and 82/182 welds
  - Alloy 82/182 reactor vessel core guide lug shell cladding
  - Alloy 600 reactor internals clevis inserts
  - Pressurizer surge, spray, safety, and relief nozzle 82/182 welds
  - U-tubes of the steam generators (SGs). The tubing of the Unit 2 SGs is Alloy 600.
  - Steam generator tubesheet cladding
  - Steam generator partition plate
  - Steam generator channel head drain
  - Steam generator tube-to-tubesheet welds (autogenous)

PWSCC of Nickel Base Alloy 600/82/182 Materials

The most significant factor that influences the PWSCC of Alloy 600/82/182 components is the service temperature. The two most significant Alloy 600/82/182 components that are bounding to the PWSCC susceptibility are the reactor vessel hot leg nozzle welds and the BMI nozzles. These are discussed below.

The Alloy 600 PWSCC susceptibility is a thermally activated process. The PWSCC susceptibility (S) is given by:

$$S = A(\phi yk)^4 \exp(-Q/RT)$$

where:

A is the material constant
$(\phi yk)^4$ is the stress factor
ϕy is the yield strength
and k is the residual stress factor
Q is the activation energy of the PWSCC process (~50,000 cals/mole)
R is the gas constant 1.987
T is the temperature in °R

For the current situation, since the only variable due to the SPU is the component service temperature, the susceptibility (S) can be expressed as:

$$S = B \exp(-Q/RT), \text{ B being a constant}$$

The change in the PWSCC susceptibility (ΔS) due to a change in the service temperature (ΔT) can be obtained by taking a differential and is given by:

$$\Delta S = B \exp(-Q/RT)\ (Q/RT2)\ \Delta T$$

or $\Delta S/S = (Q/RT2)\ \Delta T$

<u>Change in the PWSCC Susceptibility of the Alloy 82/182 Hot Leg Nozzle Weld</u>

The maximum change in the PWSCC susceptibility value (ΔS) of the hot leg nozzle weld was assessed from the maximum change in temperature ($\Delta Tmax$) due to the SPU. This value was established for Unit 1 from the data in Table 2.1.5-1 to be 1.2°F or 1.2°R. For Unit 2, this value was established from the data in Table 2.1.5-2 to be 1.8°F or 1.8°R.

From the equation above:

$$\Delta S/S = (0.08)\ (\Delta T°R)$$

where:

$\Delta S/S$ is the fractional change in the PWSCC susceptibility, and ΔT the change in the service temperature in units of Rankine.

On this basis, an increase in the PWSCC susceptibility of 10 percent was estimated for the hot leg nozzle weld as a result of the SPU for Unit 1. For Unit 2, an increase in the PWSCC susceptibility of 14 percent was estimated for the hot leg nozzle weld as a result of the SPU.

The increase in PWSCC susceptibility is not considered significant since the absolute susceptibility of this location is estimated to be very low (~10^{-11} failure probability).

Effect on the PWSCC Susceptibility of Alloy 600/82/182 BMI Penetrations for Unit 1

The SPU causes a net decrease in the core inlet temperature that corresponds to a decrease in the PWSCC susceptibility for the reactor vessel head penetrations (RVHPs) and BMI penetrations. Furthermore, since the BMIs are fabricated from PWSCC susceptible Alloy 600/82/182 material, the Unit 1 BMIs are subject to NRC Bulletin 2003-02 requiring certain inspections for the safe management of the BMI PWSCC issue. In support of this, Luminant Power performs visual inspection of BMIs every refueling outage.

Effect of SPU on the PWSCC Susceptibility of RVHPs for Unit 1

The Unit 1 reactor vessel closure head with Alloy 600/82/182 penetrations was replaced in early 2007 with a new head with Alloy 690/52/152 CRDM penetrations. Laboratory and field experience to date suggests that Alloy 690 and associated Alloy 52/152 welds are resistant to PWSCC. On this basis, the proposed uprating is not expected to have an impact on the PWSCC degradation of the Alloy 690/52/152 RVHPs.

Inspection Requirements for Replacement Heads with Alloy 690 Nozzles

CPNPP Unit 1 is a cold head plant and is now in the replacement head category.

Unit 1 will continue to monitor the industry programs and recommendations to manage the issue for the new vessel head and take appropriate actions as necessary.

Effect on the PWSCC Susceptibility of Alloy 600/82/182 Reactor Vessel Head Penetrations and BMI Penetrations for Unit 2

The industry experience over the past decade showed that the PWSCC susceptibility of the Alloy 600/82/182 outer most circle RVHPs is considered bounding to other Alloy 600 primary component locations due to the presence of high residual stresses and service temperatures at those penetration locations. Since Unit 2 is a cold head plant, the best-estimate mean fluid maximum service temperature at the RVHPs is considered to be the core inlet temperature for the purpose of the current evaluation. This value was established from the data in Table 2.1.5-2 to be 558°F, a 1.8°F decrease at SPU conditions.

The SPU causes a net decrease in the core inlet temperature, which corresponds to a decrease in the PWSCC susceptibility for the RVHPs and BMI penetrations. Furthermore, since the BMIs are fabricated from PWSCC susceptible Alloy 600/82/182 material, the Unit 2 BMIs are subject to NRC Bulletin 2003-02 requiring certain inspections for the safe management of the BMI PWSCC issue. In support of this, Luminant Power performs visual inspections of Unit 2 BMIs every refueling outage.

Thermal Aging

Thermal aging of cast stainless steel can lead to precipitation of additional phases in the ferrite and growth of existing carbides at the ferrite/austenitic boundaries that can result in loss of ductility and fracture toughness of the material. The susceptibility to thermal aging is a function of the material chemistry, aging temperature, and time at temperature.

An increase in the hot leg temperature of 1.2°F for Unit 1 and a 1.8°F increase in the hot leg temperature for Unit 2 was assessed due to the SPU. The effect of this change in the service temperature on the thermal aging is considered.

Topical report WCAP-14575 (Reference 1) indicates that thermal aging causes reduction in fracture toughness of the CASS component material and hence reduction in the critical flaw size that could lead to component failure. The impacted RCPB CASS components include primary piping and its welds, valve bodies, and pump casings. WCAP-14575 (Reference 1) proposed programs to manage the effects of thermal aging of CASS components during the period of extended operation. Any potential affect on thermal aging due to the SPU would be contained within the proposed programs of Reference 1.

2.1.5.2.4 Results

Based on the results of the assessment of the potential materials degradation issues resulting from the proposed SPU at CPNPP, it is concluded that:

- No new material degradation issues of carbon steel boric acid corrosion are expected due to the SPU water chemistry.
- The risk for PWSCC of the Alloy 600/82/182 BMI penetrations at Unit 1 does not increase due to the net decrease in the service temperature of the BMIs.
- The risk for PWSCC of the Alloy 600/82/182 RVHPs and BMI penetrations at Unit 2 does not increase due to the net decrease in the service temperature of the RVHPs and BMIs.
- The effect of a small increase in the hot leg temperature on the thermal aging of piping and welds was assessed. At CPNPP, Luminant Power follows the WOG recommended Aging Management Program to address the impact of thermal aging embrittlement on the LBB evaluations for the current license basis of operation. The SPU will not affect any changes to the Aging Management Program.
- The chemistry changes resulting from the SPU do not involve introduction of any of the contributors to SCC of austenitic stainless steel. Therefore, no material degradation is expected in the stainless steel components as a result of the SPU.

The results of the RCPB material degradation assessment showed that no new materials degradation issues will result from the proposed SPU at CPNPP. On this basis, it is concluded that the new SPU environmental conditions will not introduce any significant aging effects on

their components during the current license basis, nor will the SPU change the manner in which the component aging are managed by the Aging Management Program.

2.1.5.3 Conclusion

Luminant Power has reviewed the evaluation of the effects of the proposed SPU on the susceptibility of RCPB materials to known degradation mechanisms and concludes that the evaluation has identified appropriate degradation management programs to address the effects of changes in system operating temperature on the integrity of RCPB materials. Luminant Power further concludes that the evaluation has demonstrated that the RCPB materials will continue to be acceptable following implementation of the proposed SPU and will continue to meet the requirements of GDCs-1, -4, -14, and -31; 10 CFR Part 50, Appendix G; and 10 CFR 50.55a. Therefore, Luminant Power finds the proposed SPU acceptable with respect to RCPB materials.

2.1.5.4 Reference

1. WCAP-14575, "Licensing Renewal Evaluation: Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components," December 2000.

Table 2.1.5-1 Summary of Service Temperature Changes in the RV Hot and Cold Legs Due to the Proposed SPU – Unit 1			
Core Power Level (MWt)	**Penetration Location**	**Temperature (°F)**	**Maximum Change in the Steady-State Peak Temperature Due to Uprating (°F)**
3,458 (Current)	RV Hot Leg	619.2	
3,458 (Current)	RV Cold Leg, RVH, and BMI	559.2	
3,612 SPU	RV Hot Leg	606.2 – 620.4	1.2
3,612 SPU	RV Cold Leg, RVH, and BMI	542.2 – 558.0	-1.2

Table 2.1.5-2 Summary of Service Temperature Changes in the RV Hot and Cold Legs Due to the Proposed SPU – Unit 2			
Core Power Level (MWt)	**Penetration Location**	**Temperature (°F)**	**Maximum Change in the Steady-State Peak Temperature Due to Uprating (°F)**
3,458 (Current)	RV Hot Leg	618.6	
3,458 (Current)	RV Cold Leg, RVH, and BMI	559.8	
3,612 SPU	RV Hot Leg	606.2 – 620.4	1.8
3,612 SPU	RV Cold Leg, RVH, and BMI	542.2 – 558.0	-1.8

2.1.6 Leak Before Break

2.1.6.1 Regulatory Evaluation

Leak-before-break (LBB) analyses provide a means for eliminating from the design basis the dynamic effects of postulated pipe rupture for a piping system. Nuclear Regulatory Commission (NRC) approval of LBB analysis for a plant permits the licensee to: (1) remove protective hardware along the piping system (that is, pipe whip restraints and jet impingement barriers); and (2) redesign pipe-connected components, their supports, and their internals. The Luminant Power review of LBB covered:

- Direct pipe failure mechanisms (such as water hammer, creep damage, erosion, corrosion, fatigue, and environmental conditions)

- Indirect pipe failure mechanisms (such as seismic events, system overpressurizations, fires, flooding, missiles, and failures of structures, systems, and components (SSCs) in close proximity to the piping)

- Deterministic fracture mechanics and leak detection methods

The acceptance criteria are based on General Design Criterion (GDC)-4, insofar as it allows exclusion of dynamic effects of postulated pipe ruptures from the design basis.

Current Licensing Basis

The NRC has documented its review of LBB for the Comanche Peak Nuclear Power Plant (CPNPP) in Supplemental Safety Evaluation Reports (SSERs) 23 and 26.

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the adequacy of CPNPP design relative to conformance to:

- GDC-4, Environmental and Dynamic Effects Design Bases, is described in FSAR Section 3.1.1.4.

  The station's SSCs important to safety are designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accident (LOCA). Environmental conditions are described in FSAR Section 3.11.

  These SSCs are appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that can result from equipment failures and from events and conditions outside the nuclear power plant unit.

Details of the design, environmental testing, and construction of these SSCs are included in FSAR Chapters 3, 5, 6, 7, 8, 9, and 10. Evaluation of the performance of safety features is contained in FSAR Chapter 15.

The LBB methodology demonstrates that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping. It has been applied to CPNPP Units 1 and 2 to exclude from the design basis the dynamic effects of postulated pipe ruptures in the primary coolant piping and 10-inch and larger reactor coolant loop branch lines, as discussed in FSAR Sections 3.6B.2.5.1 and 3.6B.2.5. Implementation of this technology eliminates the need for pipe whip restraints and jet impingement barriers, respectively.

LBB analyses were performed for the CPNPP Unit 1 RCS primary loop, pressurizer surge line, residual heat removal (RHR) piping, accumulator injection nozzles, and the accumulator injection lines. The analyses for CPNPP Unit 1 are documented in WCAP-10527, dated April 1984 (Reference 1); WCAP-12248 Supplement 3, dated June 1990 (Reference 2); and CPSES-1 WHIPJET program report (Reference 3), dated April 1988 and May 1989; WCAP-12258 Supplement 2, dated August 1989 (Reference 4); and WCAP-12267, dated May 1989 (Reference 5).

LBB analyses were performed for the CPNPP Unit 2 RCS primary loop, pressurizer surge line, residual heat removal (RHR) lines, and the accumulator lines (including nozzles). The analyses for CPNPP Unit 2 are documented in WCAP-10527, dated April 1984 (Reference 1); WCAP-13100, dated December 1991 (Reference 6), WCAP-13165, dated December 1991 (Reference 7); and WCAP-13167 (Reference 8), dated January 1992.

FSAR Section 3.6.B.2.1.1 states, in part, that “The generic Leak-Before-Break technology described in NUREG-1061 Volume 3 has been applied to the CPNPP Units 1 and 2 RCS main loop piping.” FSAR Section 3.6.B.2.1.2 states in part that “Application of leak-before-break methodology justifies the elimination of postulated pipe ruptures in 10 inch and larger RCS branch lines. This analysis demonstrates the piping integrity and serves as the basis for excluding from consideration the dynamic effects associated with postulated pipe ruptures.”

2.1.6.2 Technical Evaluation

2.1.6.2.1 Introduction

The current structural design basis of the CPNPP Units 1 and 2 includes the application of LBB methodology to eliminate consideration of the dynamic effects resulting from pipe breaks in the RCS primary loop piping, pressurizer surge line, RHR piping, and the accumulator injection lines. The purpose of this section is to describe the evaluations performed to demonstrate that the elimination of these breaks from the structural design basis continues to be valid following implementation of the stretch power uprate (SPU), and that primary loop piping, pressurizer surge line, RHR piping, accumulator injection nozzles, and the accumulator injection lines comply with the requirements of GDC-4.

To demonstrate the elimination of primary loop pipe breaks pressurizer surge line, RHR piping, accumulator injection nozzles, and the accumulator injection lines, the following objectives had to be achieved:

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

These objectives were met in the current LBB analyses.

The current LBB analyses were evaluated to address the proposed SPU conditions.

2.1.6.2.2 Input Parameters, Assumptions, and Acceptance Criteria

The pipe stress analyses loadings, operating pressure, and temperature parameters for the SPU were used in the evaluation.

The parameters, which are important in the evaluation, are the piping forces, moments, normal operating temperature, and normal operating pressure. These parameters were used as input in the evaluation. The normal SPU operating temperature range and normal operating pressure conditions are provided in Licensing Report Section 1.1.

Acceptance Criteria

The LBB recommended margins are as follows:

- Margin of 10.0 on leak rate
- Margin of 2.0 on flaw size
- Margin of 1.0 on loads using faulted load combinations by absolute summation method or margin on loads of 1.4.

2.1.6.2.3 Description of Analyses and Evaluations

Westinghouse performed plant-specific LBB analyses for the CPNPP Unit 1 primary loop piping, pressurizer surge line, RHR line (with thermal stratification) and the accumulator injection nozzles. The results of the analyses were documented in WCAP-10527 (Reference 1) and WCAP-12248 Supplement 3 (Reference 2), WCAP-12258 Supplement 2 (Reference 4) and WCAP-12267 (Reference 5). Robert L. Cloud & Associates, Incorporated, performed

plant-specific LBB analyses for the CPNPP Unit 1, RHR piping, and the accumulator injection lines and the results were documented in WHIPJET program report (Reference 3). WCAP-10527, WCAP-12248 Supplement 3; WCAP-12258 Supplement 2; WCAP-12267 and WHIPJET program report analyses formed the basis for the SPU analyses. The primary loop piping, pressurizer surge line, RHR piping, and the accumulator injection lines dead weight, normal thermal expansion and stratification (for RHR and pressurizer surge line), and safe shutdown earthquake (SSE) and pressure loads were employed. The SPU normal operating temperature range and pressure were used in the evaluation.

Westinghouse performed plant-specific LBB analyses for the CPNPP Unit 2 primary loop piping, pressurizer surge line, RHR lines and the accumulator lines. The results of the analyses were documented in WCAP-10527 (Reference 1), WCAP-13100 (Reference 6), WCAP-13165 (Reference 7), and WCAP- 13167 (Reference 8). WCAP-10527, WCAP-13100, WCAP-13165, and WCAP-13167 formed the basis for the SPU LBB analyses. The primary loop piping, pressurizer surge line, RHR lines, and the accumulator lines, dead weight, normal thermal expansion and stratification (for RHR and pressurizer surge line), and safe shutdown earthquake (SSE) and pressure loads due to the SPU conditions were employed. The SPU normal operating temperature range and pressure were used in the evaluation. The evaluation results demonstrated that all the LBB recommended margins for the primary loop piping, pressurizer surge line, RHR lines, and the accumulator lines continue to be satisfied for the SPU conditions.

2.1.6.2.4 Results

The evaluation results demonstrated that all the LBB recommended margins for the primary loop piping, pressurizer surge line, RHR piping, accumulator injection nozzles and the accumulator injection lines are satisfied for the SPU conditions. The evaluation results demonstrated the following:

- Leak rate – A margin of 10.0 exists between the calculated leak rate from the leakage flaw and the leak detection capability of 1 gpm.
- Flaw size – A margin of 2.0 or more exists between the critical flaw size and the leakage flaw size.
- Loads – A margin of 1.0 on loads using faulted load combinations by absolute summation method or a margin on loads of 1.4 exists.

The evaluation results demonstrated that the LBB conclusions provided in current LBB analyses for CPNPP Units 1 and 2 remain unchanged for the SPU conditions.

It is therefore concluded that the LBB acceptance criteria continue to be satisfied for the primary loop piping, pressurizer surge line, RHR piping, accumulator injection nozzles, and the accumulator injection lines at the SPU conditions. All the recommended margins continue to be satisfied and the conclusions shown in the current LBB analyses remain valid. It was therefore concluded that the dynamic effects of primary loop pipe, pressurizer surge line, RHR piping,

accumulator injection nozzles, and the accumulator injection lines breaks, need not be considered in the structural design basis of the CPNPP Units 1 and 2 at the SPU conditions.

2.1.6.3 Conclusions

Luminant Power has reviewed the evaluation of the effects of the SPU conditions on the LBB analyses for CPNPP Units 1 and 2 and determined that the changes in the primary system pressure and temperature range and the associated effects on the LBB analyses have been adequately addressed. Luminant Power further determined that the evaluation demonstrated that the LBB analyses will continue to remain valid following implementation of the SPU and that primary loop piping, pressurizer surge line, RHR piping, accumulator injection nozzles, and the accumulator injection lines that credit LBB will continue to meet the current licensing basis requirements with respect to GDC-4. Therefore, Luminant Power finds the SPU acceptable with respect to LBB for CPNPP Units 1 and 2.

2.1.6.4 References

1. WCAP-10527, "Technical Bases for Eliminating Large Primary Loop Pipe Rupture as a Structural Design Bases for Comanche Peak Units 1 and 2," April 1984.

2. WCAP-12248 Supplement 3, "A Supplementary Assessment of Leak-Before-Break for the Pressurizer Surge Line of Comanche Peak Unit 1," June 1990.

3. CPSES-1 WHIPJET Program Report, April 1988 and May 1989 (for RHR Suction Hot Leg 12 inch diameter piping and Accumulator injection 10 inch diameter piping).

4. WCAP-12258 Supplement 2, "Evaluation of Thermal Stratification for the Comanche Peak Unit 1 Residual Heat Removal Lines," August 1989.

5. WCAP-12267, "Technical Bases for Eliminating Rupture of the Accumulator Injection Nozzles as a Structural Design Bases for Comanche Peak Unit 1," May 1989.

6. WCAP-13100, "Technical Justification for Eliminating Pressurizer Surge Line Rupture from the Structural Design Basis for Comanche Peak Unit 2," December 1991.

7. WCAP-13165, "Technical Justification for Eliminating Residual Heat Removal Lines Rupture as the Structural Design Basis for Comanche Peak Nuclear Power Plant Unit 2," December 1991.

8. WCAP-13167, "Technical Justification for Eliminating 10 inch Accumulator Lines Rupture as the Structural Design Basis for the Comanche Peak Nuclear Plant Unit 2," January 1992.

2.1.7 Protective Coating Systems (Paints) – Organic Materials and Inorganic

2.1.7.1 Regulatory Evaluation

Protective coating systems (paints) provide a means for protecting the surfaces of facilities and equipment from corrosion and contamination from radionuclides and also provide wear protection during plant operation and maintenance activities. This review covers protective coating systems used inside the containment and their suitability and stability under design basis loss-of-coolant accident (DBLOCA) conditions, considering radiation and chemical effects.

Current Licensing Basis

Qualified protective coatings systems used inside containment were qualified in accordance with American National Standards Institute (ANSI) N101.2 Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities.

2.1.7.2 Technical Evaluation

Luminant Power is in process of updating the coating qualifications as part of the resolution of Generic Safety Issue (GSI) 191. As stated in Letter logged TXX-05162 submitted to the NRC on September 1, 2005, Luminant Power is currently reevaluating declassified coatings inside containment and program changes are being made to restore a safety related coatings program and to restore qualification for containment coatings. SPU conditions will be considered as part of the restoration of the containment coating qualifications.

See Licensing Report (LR) subsection 2.6.1, Containment Functional Design, for the impact of the stretch power uprate (SPU) on DBA conditions inside containment.

2.1.7.3 Conclusion

Based on the above, it is concluded that the CPNPP protective coatings program (that is, organic and inorganic material) inside containment will continue to be acceptable following the implementation of the SPU.

2.1.8 Flow-Accelerated Corrosion

2.1.8.1 Regulatory Evaluation

Flow-accelerated corrosion (FAC) is a corrosion mechanism occurring in carbon steel components exposed to flowing single- or two-phase water. Components made from stainless steel are immune to FAC, and FAC is significantly reduced in components containing small amounts of chromium or molybdenum. The rates of material loss due to FAC depend on velocity of flow, fluid temperature, steam quality, oxygen content, and pH. During plant operation, control of these parameters is limited and the optimum conditions for minimizing FAC effects, in most cases, cannot be achieved. Loss of material by FAC will, therefore, occur. Luminant Power has reviewed the effects of the proposed stretch power uprate (SPU) on FAC and the adequacy of the FAC Program to predict the rate of loss so that repair or replacement of damaged components could be made before they reach critical thickness. It consists of predicting loss of material using the CHECWORKS computer code, visual inspection, and volumetric examination of the affected components. Luminant Power acceptance criteria are based on the structural evaluation of the minimum acceptable wall thickness for the components undergoing degradation by FAC.

Current Licensing Basis

The adequacy of the Comanche Peak Nuclear Power Plant (CPNPP) design relative to the general design criteria is discussed in FSAR Sections 3.1.1 and 3.1.2. The CPNPP FAC program is based on Generic Letter (GL) 89-08, and the guidelines in Electric Power Research Institute (EPRI) Report NSAC-202L.

Large bore pipe systems susceptible to FAC are modeled with CHECWORKS. Small bore pipe systems are not modeled; however, significant non-modeled systems are identified, examined, and tracked in CHECWORKS.

The program is designed to ensure that erosion/corrosion does not result in unacceptable degradation of the structural integrity of carbon steel piping systems. The program is documented in the CPNPP Corrosion Monitoring Program document and includes the following:

- Frequency of inspection criteria
- Acceptance criteria
- Inspection/expansion criteria
- Repair/replacement criteria
- Corrective action

The CPNPP Erosion/Corrosion Program is responsive to Generic Letter 89-08, “Erosion/Corrosion-Induced Pipe Wall Thinning,” and implements the guidelines in EPRI Report, NSAC-202L, “Recommendations for an Effective Flow-Accelerated Corrosion Program.”

2.1.8.2 Technical Evaluation

2.1.8.2.1 Introduction

The CPNPP corrosion monitoring program is designed to detect and monitor wall thinning of system components due to various forms of corrosion including FAC. Systems susceptible to FAC are modeled and analyzed with the EPRI software CHECWORKS.

The corrosion monitoring program includes guidance for evaluations of susceptible systems and components, selection of examination sites, inspections, evaluation of examination results, and corrective action strategies.

2.1.8.2.2 Description of Analyses and Evaluations

The CHECWORKS model is updated based on plant modifications or changes to plant operations/chemistry. Additionally, the CHECWORKS model is updated with new examination data after each refueling outage.

Flow-Accelerated Corrosion Susceptibility Evaluation

A susceptibility analysis for FAC has been performed for piping systems at CPNPP. Lines that are susceptible to FAC are contained in the following secondary piping systems:

- Condensate
- Extraction steam
- Feedwater
- Turbine gland system
- Heater drains
- Main steam
- Auxiliary steam
- Steam generator blowdown
- Vents and drains

Piping/Component Selection

Selection consideration for examination sites include highest relative wear, shortest relative remaining service life, components downstream of control valves, previous plant experience, industry experience, engineering judgment, and other considerations.

Inspection Techniques

Nondestructive examination (NDE) is performed as directed by the corrosion monitoring plan or responsible engineer. Examination sites are identified and grided in accordance with the corrosion monitoring plan. NDE is performed in accordance with approved procedures. NDE data reports are provided for each examination, and retained in the corrosion monitoring plan final report for each outage.

Evaluation of Inspection Data

The acceptable wall thickness is based on pipe stress calculations or an administrative limit. An administrative limit of 1/16-inch minimum wall thickness is maintained. The minimum pipe wall thickness based on pipe stress is calculated in accordance with the appropriate ASME/ANSI code.

Component Repair/Replacement

Components not meeting the acceptance criteria are documented in the CPNPP action tracking program. Repair/replacement of the component is performed in accordance with approved procedures.

Long-term strategies are reviewed prior to dispositioning unacceptable components and pipe lines. These strategies may be considered, but are not limited to:

- Component material upgrade – This would normally utilize a higher content of chromium or other FAC resistant material
- Change in configuration – Increase pipe diameter, schedule, or routing to decrease wear
- Change operation of the piping system
- Change plant chemistry to optimize pH, dissolved oxygen, or other factors
- Review other solutions utilized in the industry

CPNPP Experience Regarding Flow-Accelerated Corrosion

CPNPP has experienced minimal problems with FAC. This is attributed to excellent secondary chemistry since the commencement of commercial operation and a significant amount of pipe with high chrome content. The areas where large bore pipe wall thinning have occurred due to FAC are at five expanders in the heater drain lines downstream of control valves in Unit 1. The latest was an expander in line segment HD-1-0049 that was experiencing accelerated wear. This was identified by the FAC program and the expander was replaced in 1RF12.

SPU Conditions

Systems in the main steam, feed, heater drains, condensate, and associated drains have been reviewed to identify changes in fluid conditions (e.g., pressures, temperatures, and flow rates). These systems will experience flow increases of less than 10 percent, pressure changes of less than 10 percent, and very minor temperature changes.

However, it is noted that in the extraction steam system the extraction steam velocity to the second point heater for both Unit 1 and Unit 2 is slightly above the industry standard, but is not expected to have any impact on the integrity of the extraction steam piping system. Also, this

slight increase will not significantly impact wear rates due to flow accelerated corrosion in this line.

Second point extraction steam velocity (14" pipe section) comparisons are as follows:

- Unit 1: 4.8 percent above industry standard, but only 3.7 percent above current
- Unit 2: 5.4 percent above industry standard, but only 1.3 percent above current

Several areas that are most susceptible to flow accelerated corrosion have been reviewed. Tables 2.1.8-1 and 2.1.8-2 list five such locations for Units 1 and 2, respectively. These tables display the changes in pressure, temperature, flow velocity, and predicted wear rates from current conditions to SPU conditions. The maximum predicted change in wear rate is less than one-ten thousandth inch per year. Thus, the changes in wear rates are insignificant and SPU has minimal impact to predicted wear rates. The piping locations will continue to be monitored as part of the FAC Program.

2.1.8.3 Conclusions

The SPU has resulted in small changes in steam cycle pressures, temperatures, and flow velocities. No changes are being made in water chemistry treatment of feedwater and condensate systems for SPU conditions. From a review of Tables 2.1.8-1 and 2.1.8-2, changes in wear rate are small. The proposed SPU is acceptable with respect to FAC.

Table 2.1.8-1

Flow-Accelerated Corrosion Wear Rates for Unit 1

Line Description	Press (change) psia	Temp (change) °F	Velocity (change) ft/sec	Wear Rate (change) mils/yr
FW Heater 2A Normal Drain	13.5	3.2	0.33	-0.002
SG 3 Blowdown Line	10.6	0.9	0.46	-0.018
SG 2 Blowdown Line	10.6	0.9	0.46	-0.018
Reheater 1A Normal Drain	5.2	0.6	0.32	-0.001
FW Heater 4B Normal Drain	1.9	2.7	0.07	0.041

Note:

The above table displays the **change** in pressure, temperature, flow velocity, and predicted wear rates from current conditions to SPU conditions.

The change in the predicted wear rates has decreased in some locations while in other locations the change has increased. The CHECWORKS model considers a number of factors (existing wear rates, flows, temperatures, pressures) in its predictions for wear rates. It is the combination of these factors both the magnitude (amount) of the change and the direction (increase or decrease) of the change that influence the final CHECWORKS prediction. Overall, the change in predicted wear rates is insignificant; less than one tenthousandth inch per year.

Table 2.1.8-2 Flow-Accelerated Corrosion Wear Rates for Unit 2				
Line Description	**Press (change)**	**Temp (change)**	**Flow Velocity (change)**	**Wear Rate (change)**
	psia	**°F**	**ft/sec**	**mils/yr**
SG 2 Blowdown Line	-15.2	-1.7	0	0.035
Feedwater to SG 1	-24.6	3.5	0.76	0.004
SG 4 Steam Line to HP Turbine	-16.6	-2.0	7.03	0.004
MSR 2A Drain Line to Sep Dr Tk 2A	8.0	4.1	0.06	-0.006
FW Heater 2-2A Normal Drain	12.7	4.0	0.19	-0.009

Note:

The above table displays the **change** in pressure, temperature, flow velocity, and predicted wear rates from current conditions to SPU conditions.

The change in the predicted wear rates has decreased in some locations while in other locations the change has increased. The CHECWORKS model considers a number of factors (existing wear rates, flows, temperatures, pressures) in its predictions for wear rates. It is the combination of these factors both the magnitude (amount) of the change and the direction (increase or decrease) of the change that influence the final CHECWORKS prediction. Overall, the change in predicted wear rates is insignificant; less than one tenthousandth inch per year.

2.1.9 Steam Generator Tube In-Service Inspection

2.1.9.1 Regulatory Evaluation

Steam generator tubes constitute a large part of the reactor coolant pressure boundary (RCPB). Steam generator tube in-service inspection (ISI) provides a means for assessing the structural and leak-tight integrity of the steam generator tubes though periodic inspection and testing of critical areas and features of the tubes. The Luminant Power review in this area covered the effects of changes in differential pressures, temperatures, and flow rates resulting from the proposed stretch power uprate (SPU) on plugging limits, potential degradation mechanisms (such as flow-induced vibrations), plant-specific alternate repair criteria, and redefined inspection boundaries. The Nuclear Regulatory Commission's (NRC's) acceptance criteria for steam generator tube ISI are based on 10 CFR 50.55a requirements for periodic inspection and testing of the RCPB. Additional review guidance is contained in Comanche Peak Nuclear Power Plant (CPNPP) Technical Specification 5.5.9 for steam generator surveillance requirements, Regulatory Guide 1.121 for steam generator tube plugging limits, Generic Letter (GL) 95-03 and Bulletin 88-02 for degradation mechanisms, and Nuclear Energy Institute (NEI) 97-06 for structural and leakage performance criteria, all of which form the basis for alternate repair criteria or redefined inspection boundaries.

Current Licensing Basis

Final Safety Analysis Report (FSAR) Section 5.4.2.2 describes the steam generator in-service inspection program. Steam generator ISI is conducted in accordance with the In-Service Inspection Program document. Steam generator tubing is inspected in accordance with: (a) Recommendations in Regulatory Guide 1.83, "Inservice Inspection of Steam Generator Tubes," Revision 1, July 1975; (b) Requirements of American Society of Mechanical Engineers (ASME) Section XI (Edition and Addenda as required by 10 CFR 50.55a), Subarticle IWB-2413; and (c) CPNPP Technical Specifications Section 5.5.9. Also, the guidelines of Nuclear Energy Institute (NEI) Letter 97-06, Steam Generator Program Guidelines have been adopted at CPNPP.

2.1.9.2 Technical Evaluation

2.1.9.2.1 Introduction

CPNPP Unit 1 has the Westinghouse Δ76 steam generators (SGs) that were installed during the 2007 outage (1RF12). CPNPP Unit 2 has the Westinghouse Model D-5 steam generator currently installed. The Model D-5 steam generator will continue to be in place when the proposed SPU is implemented.

The steam generators are designed to permit in-service inspection of Class 1 and 2 components, including individual tubes. The design aspects that provide access for inspection and the proposed inspection program comply with the edition of Section XI of the ASME Code, Division 1, Rules for Inspection and Testing of Components of Light Water-Cooled Plants, required by 10 CFR 50.55a, paragraph g. Both the Δ76 and the Model D-5 steam generator

have a number of access openings that make it possible to inspect and repair or replace a component according to the techniques specified. Additional technical information regarding the steam generators may be found in Licensing Report (LR) subsection 2.2.2.5.

2.1.9.2.2 Description of Analysis and Evaluation

Steam generator process parameters will change as a result of the proposed SPU. Parameters that are expected to change include temperatures, steam pressure, steam and feedwater flows, void fraction distributions, and circulation ratio.

The process of steam generator tube ISI and integrity assessment will not change as a result of the SPU. The tube integrity assessment process begins with an assessment of potential degradation mechanisms and selection of applicable non-destructive examination (NDE) techniques that will be used during the ISI to determine if any degradation exists. After performing the ISI, a condition monitoring assessment is performed to determine if there may have been structural or leakage integrity issues during the operating interval since the previous inspection. After employing conservative growth rates, an operational assessment is performed to ensure that structural and leakage integrity performance criteria will be met during the operating interval until the next inspection. Tubes that are not projected to meet the structural and/or leakage integrity criteria are then removed from service by plugging, or repaired using an approved method.

Although the process parameter changes due to the SPU may impact the initiation and growth rates of various degradation mechanisms, these changes are considered per the above assessments, and will be incorporated in the selection of the type of NDE program action.

2.1.9.3 Conclusion

The evaluation of the effects of the proposed SPU on steam generator tube integrity concludes that the evaluation has adequately assessed the continued acceptability of the plants Technical Specifications under the proposed SPU conditions and has identified appropriate degradation management inspections to address the effects of temperature, differential pressure, and flow rates on steam generator tube integrity. Luminant Power concluded that steam generator tube integrity will continue to be maintained and will continue to meet the performance criteria in NEI 797-06 and the requirements of 10 CFR 50.55a following implementation of the proposed SPU.

2.1.10 Steam Generator Blowdown System

2.1.10.1 Regulatory Evaluation

Control of secondary-side water chemistry is important for preventing degradation of steam generator tubes. The steam generator blowdown system removes steam generator secondary-side impurities and thus, assists in maintaining acceptable secondary-side water chemistry in the steam generators. The design basis of the steam generator blowdown system includes consideration of expected and design flows for all modes of operation. The steam generator blowdown system is designed to remove particulate and dissolved impurities from the steam generator secondary side during normal operation, including anticipated operational occurrences (main condenser in-leakage and primary-to-secondary leakage).

The acceptance criteria for the steam generator blowdown system is based on General Design Criterion (GDC)-14, insofar as it requires that the reactor coolant pressure boundary (RCPB) be designed so as to have an extremely low probability of abnormal leakage, of rapidly propagating fracture, and of gross rupture.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

The adequacy of the CPNPP steam generator blowdown design is assessed relative to conformance with the following:

- GDC 14, described in FSAR Section 3.1.2.5, Reactor Coolant Pressure Boundary, which requires that the RCPB shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. This regulatory requirement is applicable to the steam generator blowdown system insofar as the system is directly connected to the secondary (steam) side of the steam generators.

- The reactor coolant system (RCS) pressure boundary is designed to accommodate the system pressures and temperatures attained under the expected modes of plant operation, including anticipated transients, and to maintain the stresses within applicable stress limits (FSAR Section 5.2, Integrity of Reactor Coolant Pressure Boundary). Also, RCPB materials and selection and fabrication techniques ensure a low probability of gross rupture of significant leakage.

- In addition to the loads imposed on the system under normal operating conditions, consideration is also given to abnormal loading conditions, such as seismic and pipe rupture, which are discussed in FSAR Section 3.6B, Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping; and FSAR Section 3.7, Seismic Design.

- The leak-before-break methodology demonstrates that the probability of rupturing primary coolant piping is extremely low under design basis conditions. It has been applied to CPNPP Units 1 and 2 to exclude from the design basis the dynamic effects of postulated ruptures in the primary coolant loop piping, as discussed in FSAR Section 3.6B.2.5.2, High Energy Piping Other Than RCS Main Loop. Containment design, emergency core cooling, and environmental qualification requirements are not influenced by this modification.

- The system is protected from overpressure by means of pressure-relieving devices as required by applicable codes.

- The RCS boundary has provisions for inspection, testing, and surveillance of critical areas to assess the structural and leaktight integrity (FSAR Section 5.2, Integrity of Reactor Coolant Pressure Boundary). For the reactor vessel, a material surveillance program conforming to applicable codes is provided (FSAR Section 5.3, Reactor Vessel).

Additional steam generator blowdown system details are provided in FSAR Sections 10.4.8, Steam Generator Blowdown System; 3.6B.1 Postulated Piping Failures in Fluid Systems Inside Containment; and 9.3.2, Process Sampling System.

2.1.10.2 Technical Evaluation

2.1.10.2.1 Introduction

The steam generator blowdown system is described in FSAR Section 10.4.8. The steam generator blowdown system design functions are:

- To blowdown fluid at a continuous rate for chemistry control of each steam generator

- To recover both the blowdown water and its heat capacity

- To provide for containment isolation and high-energy line break (HELB) isolation of blowdown lines penetrating containment

Continuous blowdown from the steam generators is used to reduce the quantities of solids that accumulate as a result of the boiling process. The steam generator blowdown is used to optimize water chemistry conditions.

2.1.10.2.2 Description of Analyses and Evaluations

The steam generator blowdown system and components were evaluated to determine if they are capable of performing their intended functions at stretch power uprate (SPU) conditions. The evaluations were conservatively performed for an analyzed nuclear steam supply system (NSSS) thermal power of 3,628 MWt. The evaluations described below compared the existing design parameters of the systems/components with the SPU condition:

- Normal and surge blowdown flow rates
- Operating and design pressures and temperatures
- Flow capacity of control valves at uprated flow conditions
- Fluid velocities at normal operating conditions and the potential for increased erosion/corrosion at SPU conditions. The erosion/corrosion monitoring program is evaluated in Licensing Report (LR) subsection 2.1.8, Flow-Accelerated Corrosion.
- Safety-related valve closure and testing requirements (containment isolation) are addressed in LR subsection 2.2.4, Safety-Related Valves and Pumps.
- The review of piping/component supports is described in LR subsection 2.2.2.2; BOP (All Non-Class 1).
- Protection against dynamic effects, including missiles, pipe whip and discharging fluids are addressed in LR subsection 2.2.1, Pipe Rupture Locations and Associated Dynamic Effects and LR subsection 2.5.1.3, Pipe Failures.

2.1.10.2.3 Results

The increased steam and feedwater flow rates at SPU conditions do not significantly affect the concentration of impurities throughout the turbine cycle and the steam generators. The steam generator chemistry is not affected by the SPU. Therefore, no changes to the steam generator blowdown flow rates or operating modes are required as a result of the SPU.

The blowdown flow for CPNPP Units 1 and 2 is administratively limited to 600 gpm. The current blowdown rates are below the maximum system capacity values and are not expected to increase as a result of the SPU.

Since the flow conditions of velocity and temperature in the steam generator blowdown piping at SPU are no worse than the original design parameters, the potential for erosion/corrosion is not expected to increase due to the SPU. Changes in expected operating conditions at the SPU are considered in the flow-accelerated corrosion (FAC) program.

At SPU conditions, the operating temperatures and pressures in the steam generators, blowdown heat exchanger, and interconnecting piping and valves will not change for Unit 1 and decrease slightly for Unit 2 (LR subsection 2.2.2.5). The existing design pressure and temperature of the steam generator blowdown system remain bounding for SPU conditions since these values are based on the no-load operating condition, which does not change at SPU. Therefore, the design conditions for the steam generator blowdown piping and components connected to the steam generators also remain bounded for SPU conditions.

The steam generator blowdown lines penetrating containment are provided with air-operated isolation valves that are designed to close for containment isolation post-accident. These lines are also provided with HELB isolation valves. The maximum blowdown flow rates and pressures experienced by these valves at SPU do not exceed the existing valve design capabilities and, therefore, these valves continue to meet their containment isolation and HELB isolation design functions.

2.1.10.3 Conclusions

The effects of the proposed SPU on the steam generator blowdown system have been evaluated for changes in system flow and impurity levels with the conclusion that they continue to meet the CPNPP current licensing basis requirements with respect to GDC-14 following implementation of the proposed SPU. No modifications to the steam generator blowdown system are required for SPU. Therefore, the proposed SPU is acceptable with respect to the steam generator blowdown system.

2.1.11 Chemical and Volume Control System

2.1.11.1 Regulatory Evaluation

The chemical and volume control system (CVCS) (including the boron recycle system (BRS)) consists of the charging, letdown and seal water system; the chemical control, purification and makeup system; and the boron thermal regeneration system. These systems provide means for:

- Maintaining water inventory via programmed level in the pressurizer, that is, maintain required water inventory in the reactor coolant system (RCS)
- Maintaining seal water flow to the reactor coolant pumps (RCPs)
- Controlling reactor coolant water chemistry conditions, activity level, soluble chemical neutron absorber concentration and makeup
- Emergency core cooling (part of the system is shared with the emergency core cooling system)
- Providing means for filling, draining, and pressure testing of the RCS

The acceptance criteria are based on:

- General Design Criterion (GDC)-14, insofar as it requires that the reactor coolant pressure boundary (RCPB) be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating fracture, and of gross rupture
- GDC-29, insofar as it requires that the reactivity control systems be designed to ensure an extremely high probability of accomplishing their safety functions in anticipation of operational occurrences

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the adequacy of CPNPP safety-related structures, systems, and components (SSCs) with respect to nuclear design relative to conformance to:

- GDC-14, Reactor Coolant Pressure Boundary, is described in FSAR Section 3.1.2.5.

  The RCS boundary is designed to accommodate the system pressures and temperatures attained under the expected modes of plant operation, including anticipated transients, and to maintain the stresses within applicable stress limits (See FSAR Section 5.2). Also, RCS pressure boundary materials, selection, and fabrication techniques ensure a low probability of gross rupture or abnormal leakage.

  In addition to the loads imposed on the system under normal operating conditions, consideration is also given to abnormal loading conditions, such as seismic and pipe rupture, as discussed in FSAR Sections 3.6 and 3.7.

  The RCS boundary has provisions for inspection, testing, and surveillance of critical areas to assess their structural and leak tight integrity (See FSAR Section 5.2).

- GDC-29, Protection Against Anticipated Operational Occurrences, is described in FSAR Section 3.1.3.10.

  The CPNPP protection and reactivity control systems are designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences. FSAR Section 7.5.3 provides details of system design.

As described in FSAR Section 9.3.4.1.1.1, the CVCS supports reactivity control, in addition to the reactivity control achieved by the control rods. Reactivity is controlled by regulating the concentration of boric acid solution, which acts as a neutron absorber, in the RCS.

As described in FSAR Section 6.2.4, the CVCS supports containment isolation system functions of limiting the release of potentially radioactive materials to the environment through CVCS piping sections that penetrate the containment.

2.1.11.2 Technical Evaluation

2.1.11.2.1 Introduction

The CVCS is described in FSAR Section 9.3.4. The system is designed to perform the following functions:

- Maintenance of programmed water level in the pressurizer, that is, maintain required water inventory in the RCS

- Maintenance of seal water injection flow to the RCPs

- Control of reactor coolant water chemistry conditions, activity level, soluble chemical neutron absorber concentration, and makeup

- Emergency core cooling (part of the system is shared with the emergency core cooling system)

- Provide means for filling, draining, and pressure testing of the RCS

To perform the functions identified above, continuous feed and bleed is maintained between the RCS and the CVCS. Borated water is letdown from the RCS, through a regenerative heat exchanger (HX), to minimize thermal loss from the RCS. The pressure is reduced through orifices and further cooling occurs in the letdown HX followed by a second pressure reduction. Borated water is returned to the RCS by the charging system, which also provides seal injection flow to the RCPs.

The RCS chemistry is maintained by passing the letdown flow through demineralizers that remove ionic impurities. A filter removes suspended solids, and the gases dissolved in the coolant can be removed via the spray nozzles in the volume control tank (VCT) while hydrogen gas is continually added to the coolant in the VCT. The boric acid concentration in the coolant is changed by the reactor makeup portion of the CVCS as required for reactivity control. Excess coolant may be diverted into the boron recovery portion of the CVCS for processing.

The CVCS also provides a means for adding chemicals to the RCS which control the pH of the coolant during initial startup and subsequent operation, scavenge oxygen from the coolant during startup, and counteract the production of oxygen in the reactor coolant due to radiolysis of water in the core region. The CVCS has the ability to maintain the RCS water chemistry within the limits specified in FSAR Table 5.2-5.

The function of soluble neutron absorber (boron) concentration control and makeup is provided by the reactor makeup control system using 4-weight-percent boric acid solution and reactor makeup water from the reactor makeup water storage tank. In addition, for emergency boration and makeup, the capability exists to provide refueling water or 4-weight-percent boric acid to the suction of the centrifugal charging pumps (CCPs).

The CCPs in the CVCS also serve as the high-head safety injection pumps in the emergency core cooling system. Other than the CCPs and associated piping and valves, the CVCS is not required to function during a loss-of-coolant accident (LOCA). During a LOCA, the CVCS is isolated except for the CCPs and the piping in the safety injection path.

The BRS is capable of processing reactor coolant to recover primary grade water and boric acid for reuse or disposal. The liquid entering the BRS is produced by the feed-and-bleed operations necessary to maintain the boron concentration in the reactor coolant at the desired level. This liquid is reactor coolant letdown fluid from the CVCS. The liquid can be processed through a mixed bed demineralizer.

The primary system recovery holdup capacity is evaluated for routine plant changes, such as core reloads, and infrequent plant changes such as a plant uprating that result in a change to core operating conditions and initial core reactivity. Boron recovery holdup capability is addressed on a cycle-specific basis.

The boron thermal regeneration system (BTRS) is designed to allow load follow operations as required by the design load cycle. Load follow operation is currently not utilized at CPNPP, therefore, the BTRS chiller and associated components in the chilled water loop are normally removed from service. The BTRS is capable of reducing reactor coolant boron concentration at the end of the core cycle.

2.1.11.2.2 Description of Analysis and Evaluations

The CVCS was evaluated to determine if the system is capable of performing its intended functions for the range of NSSS design parameters approved for LR Section 1.1, NSSS Parameters. The evaluation was performed for an analyzed NSSS thermal power of 3,628 MWt.

The changes in NSSS design parameters that could potentially affect the CVCS design bases functions include the increase in core power and the allowable range of RCS full-load design temperatures. The increase in core power and the allowable range of RCS full-load design temperature may also affect the CVCS design bases requirements related to the core re-load boron requirements. Additionally, the allowable range of RCS full-load design temperatures may affect the heat loads that the CVCS HXs must transfer to the component cooling water system (CCS) and in the case of the regenerative HX to the charging flow.

The RCS fluid interfaces with the CVCS are the regenerative, letdown, seal water, and excess letdown HXs. Design and operating conditions of the HXs are reviewed to assure that the uprating conditions are bounded by the HX design and operating conditions.

Regenerative Heat Exchanger

The regenerative HX cools the normal letdown flow from the RCS, which is at RCS T_{cold} temperature. The design inlet (RCS T_{cold}) temperature of the regenerative HX is 560°F, which bounds the highest RCS T_{cold} temperature of 558.0°F for SPU conditions (LR Section 1.1). Charging flow and temperature remains the same at SPU conditions. Since the inlet (letdown) temperature at uprating conditions (558.0°F) is lower than the design inlet temperature from the HX spec sheets, the outlet temperature (charging) is not adversely impacted by the uprating. This results in a lower inlet temperature to the letdown HX that is less than the design process inlet temperature of 290°F. The letdown (shell side) design temperature is 650°F, which bounds uprating conditions from a mechanical design standpoint. Therefore, the performance of the regenerative HX remains essentially unchanged due to the stretch power uprate (SPU) and is acceptable with no plant changes required.

Letdown Heat Exchanger

The letdown HX cools the letdown flow from the regenerative HX. Since the performance of the regenerative HX is essentially unchanged at SPU conditions, as discussed in the previous section, there is essentially no effect on the performance of the letdown HX. The normal inlet temperature will remain at 290°F. The letdown (tube side) design temperature of 400°F exceeds the original operating inlet temperature of 380°F, which also bounds the SPU conditions from a mechanical design standpoint. Therefore, it is concluded that acceptable letdown HX performance is provided at SPU conditions, with no plant changes required.

Excess Letdown Heat Exchanger

The excess letdown HX cools the excess letdown flow from the RCS cold leg Loop 1. An inlet (RCS T_{cold}) temperature of 560.0°F was analyzed for the excess letdown HX, which bounds the highest RCS T_{cold} temperature associated with the RCS T_{avg} window for the SPU. In addition, the letdown (tube side) design temperature of 650°F bounds the SPU conditions of 558°F for RCS T_{cold} from a mechanical design standpoint. The performance of the excess letdown HX is acceptable with no plant changes required.

Seal Water Heat Exchanger

The seal water HX cools the seal return flow from the RCP seal water return to the VCT, reactor coolant discharged from the excess letdown HX (if in service), and the miniflow from a CCP. The RCP heat load (including thermal barrier HX) is a function of RCS T_{cold} temperature, while the excess letdown heat load is a function of excess letdown HX performance, and the miniflow heat load is a function of the letdown HX performance. Since the SPU RCS T_{cold} temperature remains below design conditions of the letdown HX and excess letdown HX, the performance of the seal water HX is acceptable at the SPU conditions with no plant changes required.

Charging, Letdown, and RCS Makeup (Boration, Dilution, Purification, and N-16 Delay Time)

As discussed in the previous sections for the various CVCS HXs, there are essentially no effects on their performance at the SPU conditions. The charging and letdown flows are not impacted by the SPU since the RCS pressure and the CVCS orifice alignment remain unchanged.

The flow capacity performance of the RCS makeup system is independent of the change in RCS conditions resulting from the SPU conditions. However, the makeup system also relies on storage capacity of various sources of water including primary makeup water, from the reactor makeup water storage tank, and boric acid solutions from both the boric acid storage tanks and the refueling water storage tank (RWST).

Primary makeup water is used to dilute RCS boron, to provide positive reactivity control, or to blend concentrated boric acid to match the prevailing RCS boron concentration during RCS inventory makeup operations. Since the flow capacity performance of the RCS makeup system is independent of the change in RCS conditions resulting from the SPU conditions as discussed

above, the SPU does not affect the capability of the makeup water system to perform these system functions.

The boric acid storage tanks (BATs) and RWST provide the sources of boric acid for providing negative reactivity control to supplement the reactor control rods. The SPU is expected to have a small effect on the boration requirements that must be provided by the CVCS boration capabilities. The maximum expected RCS boron concentrations are within the capability of the CVCS. The boron capability is evaluated for routine plant changes, such as core reloads, and infrequent plant changes such as a plant uprating that result in a change to core operating conditions and initial core reactivity. Boron capability is addressed on a cycle-specific basis.

The CVCS letdown flow is fixed and charging flow is varied to control pressurizer water level and RCS inventory. The pressurizer water level is programmed as a function of temperature to accommodate RCS coolant expansion. Accordingly, this programmed level is being changed based on the SPU NSSS design parameters. However, this change has no impact on the ability of the CVCS to maintain RCS inventory, which is accomplished via letdown, charging, and makeup.

The letdown flow path is routed inside containment such that there is adequate decay of N-16 before the letdown fluid leaves the Containment Building. It is noted that the letdown line and excess letdown line radiation dose rates from N-16 (for example, amount of N-16) will increase proportional to the increase in reactor power level. Since the letdown flow rate is essentially unchanged, as discussed in the previous paragraphs, this radiation protection feature of the CVCS is not impacted by the SPU.

Results

The evaluations of the CVCS charging, letdown, and RCS makeup performance show the CVCS is acceptable at the SPU conditions, with no plant changes. Accordingly, the performance of the following CVCS functions (which are accomplished via charging, letdown, and makeup) are acceptable at the SPU conditions with no plant changes:

- Maintenance of programmed water level in the pressurizer, that is, maintain required water inventory in the RCS
- Maintenance of seal water injection flow to the RCPs
- Control of reactor coolant water chemistry conditions, activity level, soluble chemical neutron absorber concentration, and makeup
- Emergency core cooling (part of the system is shared with the emergency core cooling system)
- Provide means for filling, draining, and pressure testing of the RCS

The CVCS boration capability was performed for the SPU. The boron requirements were determined to be acceptable with margin remaining in the flow, volume, and time limits.

The performance of the CVCS components including valves and piping that support containment isolation are not affected by change in RCS design parameters resulting from the SPU. The requirement for containment isolation is not impacted.

There is a small increase in letdown line dose rates from N-16, proportional to the increase in reactor power level. This small increase has been evaluated in LR subsection 2.10.1, Occupational and Public Radiation Doses.

The CVCS support functions provided by the waste disposal system are not affected by the change in RCS conditions resulting from the SPU.

2.1.11.3 Conclusions

Luminant Power has reviewed the evaluations of the effects of the SPU on the CVCS and BRS and concludes that the CVCS and BRS are not impacted by changes in the temperature of the reactor coolant. It is further concluded that the CVCS and boron recycle system continue to be acceptable and continue to meet the requirements of GDC-14 and -29 following implementation of the SPU. Therefore, Luminant Power finds the SPU acceptable with respect to the CVCS.

2.2 MECHANICAL AND CIVIL ENGINEERING

2.2.1 Pipe Rupture Locations and Associated Dynamic Effects

2.2.1.1 Regulatory Evaluation

Safety-related structures, systems, and components (SSCs) important to safety could be impacted by the pipe-whip dynamic effects of a pipe rupture. Luminant Power conducted a review of pipe rupture analyses to ensure that SSCs important to safety are adequately protected from the effects of pipe ruptures. The review covered: (1) the implementation of criteria for defining pipe break and crack locations and configurations; (2) the implementation of criteria dealing with special features, such as augmented in-service inspection programs or the use of special protective devices such as pipe-whip restraints; (3) pipe-whip dynamic analyses and results, including the jet thrust and impingement forcing functions and pipe-whip dynamic effects; and (4) the design adequacy of supports for SSCs to ensure that the intended design functions of the SSCs will not be impaired to an unacceptable level as a result of pipe-whip or jet impingement loadings. The review focused on the effects that the proposed stretch power uprate (SPU) may have on items (1) through (4) above.

The acceptance criteria for the review of pipe rupture location and associated dynamic effects are based on:

- General Design Criterion (GDC)-4, insofar as it requires SSCs important to safety be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents (LOCAs). These SSCs shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

The adequacy of the CPNPP design is assessed relative to conformance to:

- GDC-4, as described in FSAR Section 3.1.1.4, Environmental and Dynamic Effects Design Bases (Criterion 4), which states that SSCs important to safety shall be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operating, maintenance, testing, and postulated accidents, including LOCAs. These SSCs shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

The environmental effects due to pipe breaks are addressed in LR Section 2.3.1.

Conformance to the requirements of GDC-4, ensuring that safety-related SSCs are adequately protected with respect to pipe ruptures and their associated dynamic effects, is discussed in FSAR Section 3.6B.

FSAR Section 3.6B.1 sets forth the design bases, description, and safety evaluation for protection and determination of the dynamic effects associated with the postulated rupture of piping in fluid systems both inside and outside containment.

FSAR Section 3.6B.2 describes the design bases for locating postulated breaks and cracks in piping inside and outside of the containment, the procedure used to define the jet thrust reaction at the break or crack location and the impingement loading on adjacent structures, equipment, systems and components.

The leak-before-break (LBB) methodology described in NUREG-1061 Volume 3 demonstrates that the probability of reactor coolant system (RCS) main loop piping breaks is extremely low under design basis conditions. It has been applied to CPNPP to exclude the dynamic effects of postulated pipe ruptures in the primary coolant loop piping and 10-inch and larger reactor coolant loop branch lines from the design basis, as discussed in FSAR Sections 3.6B.2.1.1 and 3.6B.2.1.2. Containment design, emergency core cooling and environmental qualification requirements are not influenced by the implementation of the SPU.

2.2.1.2 Technical Evaluation

2.2.1.2.1 Introduction

SSCs could be impacted by the pipe-whip dynamic effects of a pipe rupture. A review of pipe rupture analyses was conducted to ensure that those SSCs are adequately protected from the effects of pipe ruptures. The review covered:

- The implementation of criteria for defining pipe break and crack locations and configurations

- The implementation of criteria dealing with special features, such as augmented in-service inspection programs or the use of special protective devices such as pipe-whip restraints

- Pipe-whip dynamic analyses and results, including the jet thrust and impingement forcing functions and pipe-whip dynamic effects

- The design adequacy of supports for SSCs provided to ensure that the intended design functions of the SSCs will remain acceptable as a result of pipe-whip or jet impingement loadings

The review focused on the effects that the proposed SPU may have on the above items.

Refer to Licensing Report (LR) subsection 2.5.1.3, Pipe Failures, for discussion of plant design for protection from piping failures outside containment.

2.2.1.2.2 Description of Analyses and Evaluations

Affected piping systems were evaluated to address revised SPU operating conditions.

The criterion for defining pipe break and crack locations is unaffected by the SPU.

The criterion for dealing with special features, such as augmented in-service inspection programs or the use of special protective devices such as pipe whip restraints, is unaffected by the SPU.

With respect to applicable pipe rupture postulation criteria, these criteria were reviewed and changes to existing piping system stress levels resulting from the SPU were reconciled against these documents. The evaluations performed for these piping systems did not result in any new or revised pipe break locations. The evaluations performed for these systems, with the exception of the main feedwater system, did not identify any significant increases in operating conditions that would impact existing design basis pipe break, jet impingement, and pipe-whip analyses. Hence, for these systems with the exception of the main feedwater system, the current design basis analyses and design adequacy for supports for pipe break, jet impingement, and pipe-whip considerations remains valid for the SPU conditions.

The feedwater system will experience an increase in operating pressure due to SPU. Any resulting modifications to existing pipe whip restraints and/or pipe supports will be provided, if required, to accommodate the higher pipe break loadings.

The applicable piping loads resulting from the SPU conditions, as defined by LR Section 1.1, were evaluated and confirmed for continued applicability of LBB (Refer to LR subsection 2.1.6, Leak Before Break).

2.2.1.2.3 Results

Based on the evaluations performed for the SPU noted above, the following were demonstrated:

- Existing criterion for defining pipe break and crack locations and configurations is unaffected by the SPU. No new or revised pipe break locations have resulted due to SPU conditions.

- Criterion dealing with special features, such as augmented in-service inspection programs or the use of special protective devices, such as pipe-whip restraints, is unaffected by the SPU.

- Pipe-whip dynamic analyses and results, including the jet thrust and impingement forcing functions and pipe-whip dynamic effects, will remain valid for the SPU.

- The final design of SSCs remain acceptable to protect safety-related SSCs from the effects of pipe-whip and jet impingement loading for the SPU.

Hence, the design features for the CPNPP that protect safety-related SSCs from the consequences of postulated piping failures both inside and outside containment as described in FSAR Section 3.6B remain valid for SPU.

2.2.1.3 Conclusion

The evaluations related to determinations of rupture locations and associated dynamic effects have adequately addressed the effects of the proposed SPU. The evaluations have demonstrated that SSCs important to safety will continue to meet the requirements of GDC-4 following implementation of the proposed SPU. Therefore, Luminant Power finds the proposed SPU acceptable with respect to the determination of rupture locations and dynamic effects associated with the postulated rupture of piping.

2.2.2 Pressure-Retaining Components and Component Supports

The following subsection generally applies to all the specific nuclear steam supply system (NSSS) components addressed individually in later Technical Evaluation subsections. In addition to the Regulatory Evaluation, any amplifications or qualifications necessary for individual component types are provided in the Introduction section for each component.

Regulatory Evaluation

Luminant Power has reviewed the structural integrity of pressure-retaining components (and their supports) designed in accordance with the American Society of Mechanical Engineers Boiler and Pressure Vessel (ASME B&PV) Code, Section III, Division 1 and General Design Criteria (GDC) -1, -2, -4, -14, and -15. The review focused on the effects of the proposed stretch power uprate (SPU) on the design input parameters and the design basis loads and load combinations for normal, upset, emergency, and faulted conditions. The review also included a comparison of the resulting stresses and cumulative usage factors (CUFs) against Code-allowable limits.

The acceptance criteria are based on:

- 10 CFR 50 Part 55a and GDC-1, insofar as they require that structures, systems, and components (SSCs) important to safety be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed.
- GDC-2, insofar as it requires that SSCs important to safety be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions.
- GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.
- GDC-14, insofar as it requires that the reactor coolant pressure boundary (RCPB) be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating fracture.
- GDC-15, insofar as it requires that the reactor coolant system (RCS) be designed with sufficient margin to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the pressure retaining components and component supports' design adequacy regarding conformance to:

- 10 CFR Part 50.55a is described in FSAR Section 5.2.1.1, Compliance with 10 CFR Part 50.55a.

  RCS components are designed and fabricated in accordance with 10 CFR Part 50.55a. The actual addenda of the ASME B&PV Code applied to the original design of each component are listed in FSAR Table 5.2-1.

- GDC-1 is described in the FSAR Section 3.1.1.1, General Design Criteria 1 – Quality Standards and Records.

  The systems and components of the facility are classified according to their importance in the prevention and mitigation of accidents. Reactor components use the classification system developed by American National Standards Institute (ANSI) N18.2-1973, Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants. Classifications and any deviations are described in FSAR Section 3.2. Each component is given a safety class designation.

  FSAR Chapter 17 provides direct reference to the Quality Assurance Program established to provide assurance that safety-related SSCs satisfactorily perform their intended safety functions. The procedures for generating and maintaining appropriate design, fabrication, erection, and testing records are contained within the referenced documents.

- GDC-2 is described in the FSAR Section 3.1.1.2, General Design Criteria 2 – Design Bases for Protection Against Natural Phenomena.

  The natural phenomena and their magnitude are selected in accordance with their probability of occurrence at the CPNPP site. The design is based on the most severe of the natural phenomena recorded for the site, with an appropriate margin to account for uncertainties in the historical data. The natural phenomena postulated in the design are presented in FSAR Sections 2.3, 2.4, 2.5, and 3.3. The design criteria for the SSCs affected by each natural phenomenon are presented in FSAR Sections 3.2, 3.3, 3.4, 3.5, 3.7, and 3.8.

  Combinations of natural phenomena and plant originated accidents considered in the design are identified in FSAR Sections 3.8, 3.9, and 3.10. The importance of the safety functions is identified with the classification system developed by the American Nuclear Society (ANS) and is generally in accordance with ANSI N18.2-1973, Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants. This identification and any deviations are included in FSAR Section 3.2.

- GDC-4 is described in the FSAR Section 3.1.1.4, General Design Criteria 4 – Environmental and Missile Design Bases.

  The station's SSCs important to safety are designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCA. Environmental conditions are described in FSAR Section 3.11.

  These SSCs are appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that can result from equipment failures and from events and conditions outside the nuclear power unit.

  Details of the design, environmental testing, and construction of these SSCs are included in FSAR Chapters 3, 5, 6, 7, 8, 9, and 10. Evaluation of the performance of safety features is contained in FSAR Chapter 15.

  The leak-before-break (LBB) methodology demonstrates that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping (see LR Section 2.1.6). LBB methodology has been applied to CPNPP Units 1 and 2 to exclude from the design basis the dynamic effects of postulated pipe ruptures in the primary coolant loop piping and 10 inch and larger reactor coolant loop branch lines, as discussed in FSAR Section 3.6B.2.5.1 and 3.6B.2.5. Implementation of this technology eliminates the need for pipe whip restraints and jet impingement barriers, respectively. Containment design, emergency core cooling, and environmental qualification requirements are not influenced by the application of LBB methodology.

- GDC-14 is described in FSAR Section 3.1.14, General Design Criteria 14 – Reactor Coolant Pressure Boundary.

  The RCS pressure boundary is designed to accommodate the system pressures and temperatures attained under the expected modes of plant operation, including anticipated transients, and to maintain the stresses within applicable stress limits (FSAR Section 5.2). Also, RCPB materials and selection and fabrication techniques ensure a low probability of gross rupture of significant leakage.

  In addition to the loads imposed on the system under normal operating conditions, consideration is also given to abnormal loading conditions, such as seismic and pipe rupture, which are discussed in FSAR Sections 3.6 and 3.7.

  LBB methodology demonstrates that the probability of rupturing primary coolant piping is extremely low under design basis conditions. LBB methodology has been applied to CPNPP Units 1 and 2 to exclude from the design basis the dynamic effects of postulated ruptures in the primary coolant loop piping, as discussed in FSAR Section 3.6B.2.5.1. Implementation of this technology eliminates the need for primary coolant loop piping whip restraints and jet impingement barriers. Containment design, emergency core

cooling, and environmental qualification requirements are not influenced by the application of LBB methodology.

- GDC-15 is described in FSAR Section 3.1.2.15, General Design Criteria 15 – Reactor Coolant System Design.

  The design pressure and temperature for each component in the RCS and associated auxiliary, control, and protection systems are selected to be above the maximum coolant pressure and temperature under all normal and anticipated transient load conditions.

  In addition, RCPB components achieve a large margin of safety by the use of proven ASME materials and design codes, use of proven fabrication techniques, nondestructive shop testing, and of integrated hydrostatic testing of assembled components.

  FSAR Chapter 5 discusses the RCS design.

2.2.2.1 NSSS Piping, Components, and Supports

2.2.2.1.1 Regulatory Evaluation

The current licensing bases contained in Licensing Report (LR) subsection 2.2.2 apply.

2.2.2.1.2 Technical Evaluation

2.2.2.1.2.1 Introduction

The nuclear steam supply system (NSSS) piping, which is the reactor coolant system (RCS) piping, consists of four similar heat transfer reactor coolant loops (RCLs) connected in parallel to the reactor pressure vessel (RPV). Each RCL contains a Model 93A reactor coolant pump (RCP) and a Model Δ76 steam generator for Unit 1 and a D-5 steam generator for Unit 2. Each RCS loop consists of three legs: the hot leg from the RPV to the steam generator, the cross-over leg from the steam generator to the RCP, and the cold leg from the RCP to the RPV. The system also includes a pressurizer, pressurizer relief tank, connecting piping, and the instrumentation for operational control. The pressurizer is connected to Loop 4. Auxiliary system piping connections into the RCS piping are provided as necessary. The RCS piping system is supported by the primary equipment supports of the RCS, namely the RPV supports, the steam generator supports, the RCP supports, and the pressurizer supports.

As described in Chapter 5 of the Final Safety Analysis Report (FSAR), the NSSS piping, components, and supports are evaluated for the Comanche Peak Nuclear Power Plant (CPNPP) Units 1 and 2 SPU Program. The parameters associated with the SPU program were reviewed for impact on the recently performed RCL structural analysis with Δ76 steam generators, RCL primary equipment support loads, and pressurizer surge line evaluation including thermal stratification for the following:

- RCL loss-of-coolant accident (LOCA) analysis using loop LOCA hydraulic forces for the SPU program and the associated loop LOCA RPV motions for the SPU program
- RCL piping stresses
- RCL displacements at auxiliary piping line connections to the centerline of the RCL at branch nozzle connections and impact on the auxiliary piping systems
- Primary equipment nozzle loads
- RCL piping system leak-before-break (LBB) loads for LBB evaluation
- Pressurizer surge line piping analysis including the effects of thermal stratification
- RCL primary equipment support loads (reactor vessel, steam generator, and RCP)

2.2.2.1.2.2 Input Parameters, Assumptions, and Acceptance Criteria (Unit 1)

The following five basic sets of input parameters were considered in the evaluation for the SPU:

- NSSS Performance Capability Working Group (PCWG) design parameters (LR Section 1.1)
- NSSS design transients (LR Section 2.2.6)
- LOCA hydraulic forcing functions (HFFs) loads and associated RPV motions
- Jet impingement and subcompartment pressurization loads
- Feedwater, main steam line, and auxiliary feedwater line breaks

The acceptance criteria for the RCL analysis, including the RCL branch nozzles remain unchanged from the design basis analysis. For stress analysis purposes, the RCL piping is qualified in accordance with the requirements established in the American Society of Mechanical Engineers (ASME) Code 1977 Edition including Summer 1979 (Reference 1).

Class 1 Auxiliary lines are qualified in accordance with the ASME Code, Section III, 1977 Edition including Summer 1979 (Reference 1). The acceptance criteria for the pressurizer surge line

thermal stratification analysis is per the ASME Boiler and Pressure Vessel (B&PV) Code (Reference 2), and are as specified in the current design basis.

The pressurizer surge line was evaluated to the ASME B&PV Section III, Subsection NB 1986 Code (Reference 2), and includes the fatigue evaluation and the effects of thermal stratification.

The parameters associated with the CPNPP Units 1 and 2 SPU Program were reviewed for impact on the existing RCL piping and subsequent impact to the auxiliary lines attached to the RCL centerline at the RCL branch nozzle connections. The conclusions of this review are summarized below.

Nuclear Steam Supply System Performance Capability Working Group Design Parameters (Unit 1)

The PCWG design parameters as identified in LR Section 1.1 for the 3,628 MWt NSSS power were used in the thermal analysis of the RCL and used in the evaluation for the pressurizer surge line. The RCL was reconciled for two temperature cases – one for the lower bound temperature case (Cases 1 and 2), and the second for the upper bound temperature case (Cases 3 and 4), as identified in LR Section 1.1.

The RCL piping in the recently performed analysis for the Δ76 steam generator was evaluated for the conditions associated with RCS for two temperature cases: the upper bounding case with a hot leg temperature of 620.1°F, cross-over leg temperature of 558.9°F, and a cold leg temperature of 559.2°F; and the lower bound temperature case with a hot leg temperature of 605.0°F, cross-over leg temperature of 542.2°F, and a cold leg temperature of 542.6°F. The RCL upper bound temperatures for the SPU increase by 0.3°F for the hot leg, decrease by 1.3°F for the cross-over leg, and decrease by 1.2°F for the cold leg as compared to the current upper bound case temperatures. The RCL lower bound temperatures for the SPU increase by 1.2°F for the hot leg, decrease by 0.3°F for the cross-over leg, and decrease by 0.4°F for the cold leg as compared to the current lower bound case temperatures. Since the maximum increase/decrease in the RCL upper bound/lower bound temperatures of the hot leg, cross-over leg, and cold leg is only 1.3°F or less in comparison to the hot leg, cross-over leg, and cold leg temperatures in the current RCL thermal analysis, the changes in the temperatures due to the SPU have an insignificant impact on the current RCL thermal analysis. Therefore, the piping stresses, primary equipment nozzle loads and primary equipment support loads, currently qualified, are still applicable for the SPU program.

The impact on the LBB loads at the RCL weld locations generated for the current analysis of record due to the SPU program was evaluated and found insignificant. Therefore, the CPNPP Unit 1 LBB loads primary loop piping "high normal" and "low normal" operating thermal loads for the 4 loops from the current analysis are bounding and still valid for the SPU program.

The potential for stratification in the pressurizer surge line is increased as the difference in temperature between the pressurizer and the hot leg increases. The controlling ΔTs for the pressurizer surge line are associated with the plant heatup and cooldown events that are not affected by the SPU program, and the temperature changes of 10.8°F due to the SPU program

has an insignificant impact on the usage factor – ASME Code Section III equations 12 and 13 qualification. Therefore, the SPU has no adverse impact on either the thermal stratification or the fatigue analysis for the pressurizer surge line, and the fatigue results remain valid. The effect on the surge piping thermal loads at the weld locations for the temperature changes of 10.8°F due to the SPU program has been evaluated and new set of LBB loads were provided for evaluation and confirmation of LBB (LR subsection 2.1.6).

NSSS Design Transients (Unit 1)

The impact on design transients due to the changes in full-power operating temperatures for the SPU program is addressed in LR subsection 2.2.6.

There are no changes to the NSSS design transients for CPNPP Unit 1 due to the SPU that impact this evaluation. Also, the auxiliary line transients, including pressurizer surge line transients, do not change. Therefore, there is no significant impact on the piping for the SPU due to the NSSS design transients and no adverse effect on the current fatigue evaluation of the reactor coolant loop piping, and the RCL auxiliary branch nozzle fatigue evaluation. The SPU program has no adverse effect on either the thermal stratification or the fatigue analysis for the pressurizer surge line and the current results remain valid.

Loop LOCA Hydraulic Forcing Functions Forces and Associated Loop LOCA RPV Motions (Unit 1)

The impact on the loop LOCA HFFs due to the SPU program is addressed in LR subsection 2.8.5.6.3.6 and the associated loop LOCA RPV motions are addressed in LR subsection 2.2.3. LBB is applicable for the RCL main loop piping and the pressurizer surge line (LR subsection 2.1.6), and also for the accumulator and residual heat removal (RHR) lines. The application of LBB criteria exempted these three large diameter RCS pipe breaks from consideration.

With the application of LBB criteria, the LOCA cases and the associated RPV motions from the 4-inch pressurizer spray line break on the cold leg and the 6-inch safety injection line break on the hot leg were evaluated for the LOCA analyses for the current Analysis of Record (AOR). To account for the effects of a possible 3-inch break on the cold leg, the 4-inch pressurizer spray line break was evaluated with the jet/thrust load resulting from the break being applied at the appropriate location and orientation of either the 4-inch pressurizer spray line nozzle, the 3-inch boron injection nozzle, or the 3-inch charging line nozzle, and the results were maximized. It has been shown that the loop LOCA HFF forces and the associated loop LOCA RPV motion for the SPU program are bounded by the corresponding loop LOCA forces and RPV motions used in the current LOCA analyses. Therefore, the RCL piping system LOCA analyses for the SPU program are bounded by the RCL piping LOCA analyses and the current RCL piping LOCA analyses evaluation performed remains applicable for the SPU program.

The resultant asymmetric external pressure loads on the RCP and steam generator resulting from a postulated pipe rupture and pressure buildup in the loop compartments are contained in the current AOR. They are not affected by the power uprating, because the revised conditions

do not require any changes to the subcompartment forces that are used to calculate differential pressure loads.

Description of Analyses and Evaluations (Unit 1)

Since there are no changes in the weight of the system, deadweight analysis is not required for the CPNPP Units 1 and 2 SPU Program. The results from the current time-history seismic analysis are not impacted either. This is based on the assumption that the integrity of the support gaps is maintained such that the existing design seismic time-history analysis remains valid. Therefore, there are no changes to the RCL deadweight and seismic analysis for the SPU program.

As previously discussed, there are no significant changes to the current RCL thermal analysis due to the SPU.

Since the LOCA HFF forces and associated loop LOCA RPV motions for the Replacement Steam Generator and Snubber Elimination Program remain unchanged, and the subcompartment forces and the thrust and jet impingement forces are not impacted by the uprating, the RCL LOCA and pipe break analyses for the current design basis remains applicable for the SPU program.

In addition, the impact on the RCL piping analyses due to the secondary side breaks due to the power uprating on the existing piping analysis for the following secondary side breaks: a main steam line break at the steam generator nozzle, a main steam line break at the containment penetration anchor, a main feedwater line break at the steam generator nozzle, and an auxiliary feedwater line break at the steam generator nozzle, was evaluated and found insignificant.

Additionally, as previously discussed, the current design basis pressurizer surge line analysis results including the effects of thermal stratification were reviewed and found still applicable for the SPU program.

The impact on RCL piping displacements at the RCL branch nozzles is insignificant due to the SPU parameters, revised LOCA hydraulics, and RPV displacements. Therefore, the impact on the RCL branch nozzles due to the power uprating is considered negligible. The RCL piping and equipment displacements from the deadweight, thermal, seismic, LOCA, main steam nozzle break, feedwater nozzle break, and auxiliary feedwater line break analyses for the current analysis remain valid.

NSSS Piping, Components, and Supports Results (Unit 1)

Based on the evaluations performed for the SPU program NSSS PCWG design parameters, NSSS design transients, loop LOCA HFFs, and associated RPV motions, it is concluded that there is no adverse effect on the CPNPP Unit 1 current design basis RCL piping analyses and the current RCL piping design basis results remain acceptable for the CPNPP Units 1 and 2 SPU program.

With the continued applicability of the design transients and the insignificant changes due to the thermal analysis, the impact on the fatigue evaluation of the RCL piping and branch line nozzles is negligible. Therefore, the existing design basis fatigue analysis of the RCL piping and auxiliary line branch nozzles remains applicable for the SPU program.

Also, since there is no significant impact on the CPNPP Unit 1 RCL analyses due to the SPU program, there are no changes to the RCL displacements at the connections of the Class 1 and Non-Class 1 auxiliary lines to the RCL, no changes to the primary equipment nozzle qualification (RPV inlet and outlet nozzles, steam generator inlet and outlet nozzles, and RCP suction and discharge nozzles), no changes to the branch nozzle qualification of the auxiliary lines connected to the RCL, and no changes to the primary equipment support loads (RPV nozzle supports, steam generator columns and lateral bumpers, and RCP columns and tie rods). Therefore, the existing design basis results of the RCL piping analysis, the pressurizer surge line, including the effects of thermal stratification, auxiliary line branch nozzle loads, primary equipment nozzle loads, and primary equipment support loads, all remain applicable for the SPU program.

The maximum RCL piping stresses for the RCL piping and the corresponding Code-allowable stress values are presented in Table 2.2.2-1 for the SPU program and are the same as in the recently performed RCL analysis for the Replacement Steam Generator and Snubber Elimination Program. From the results tabulated in Table 2.2.2-1, it can be concluded that the RCL piping stresses are within the allowable limits and meet the acceptance criteria (Reference 1) and are acceptable for the SPU program.

LBB piping loads at the RCL weld locations from recently performed RCL piping analysis for the Replacement Steam Generator and Snubber Elimination Program remain bounding for the SPU program. The applicable thermal LBB loads, resulting from the temperature changes of 10.8°F, at the surge line weld locations due to the SPU program were provided for evaluation and confirmation of LBB. All other LBB piping loads (pressure, deadweight, and seismic loads) at the surge line welds from the existing design basis remain applicable to the SPU program.

The parameters associated with the SPU program have been evaluated for the following:

- RCL piping stresses
- RCL piping system LBB loads for LBB evaluation
- RCL piping displacements at the junction of the centerline of the RCL piping and the branch nozzle connections of the auxiliary piping systems to the RCL and impact on auxiliary piping systems
- Primary equipment nozzle loads
- Pressurizer surge line piping analysis including the effects of thermal stratification
- Primary equipment support loads (reactor vessel, steam generator, and RCP)

The evaluation indicated that the parameters associated with the SPU program have no adverse effect on the analysis of the RCL piping system, including impacts to the primary equipment nozzles. RCL piping stresses meet the required stress criteria as summarized in Table 2.2.2-1. The primary equipment nozzle loads are all acceptable. RCL piping loads transmitted for LBB evaluation for the SPU program are evaluated in LR subsection 2.1.6. The RCL primary equipment support loads are evaluated to meet the required stress criteria as summarized in LR subsection 2.2.2.3 – Reactor Vessel and Supports, subsection 2.2.2.5 – Steam Generator and Supports, and subsection 2.2.2.6 – Reactor Coolant Pumps and Supports.

RCL piping displacements at branch nozzles due to the SPU had no significant impact on the auxiliary piping systems that are attached to the RCL (as applicable) and are not affected by the SPU program.

Additionally, the current design basis analysis results for the pressurizer surge line, including the effects of thermal stratification, are still applicable and remain valid for the SPU program.

2.2.2.1.2.3 Input Parameters, Assumptions, and Acceptance Criteria (Unit 2)

The following five basic sets of input parameters were considered in the evaluation for the SPU:

- NSSS PCWG design parameters (LR Section 1.1)
- NSSS design transients (LR subsection 2.2.6)
- LOCA HFFs and associated RPV motions
- Jet impingement and subcompartment pressurization loads
- Feedwater, main steam line, auxiliary feedwater line breaks

The acceptance criteria for the RCL analysis, including the RCL branch nozzles, remain unchanged from the design basis analysis. For stress analysis purposes, the RCL piping is qualified in accordance with the requirements established in ASME Code 1977 Edition including Summer 1979 (Reference 1).

Class 1 Auxiliary lines are qualified in accordance with the ASME Code, Section III, 1977 Edition including Summer 1979 (Reference 1). The acceptance criteria for the pressurizer surge line thermal stratification analysis is per the ASME B&PV Code (Reference 2).

The pressurizer surge line was evaluated to the ASME B&PV Section III, Subsection NB 1986 Code (Reference 2), and includes the fatigue evaluation and the effects of thermal stratification.

The parameters associated with the SPU program were reviewed for impact on the existing RCL piping and subsequent impact to the RCL branch nozzle connections. The conclusions of this review are summarized below.

Nuclear Steam Supply System Performance Capability Working Group Design Parameters (Unit 2)

The PCWG design parameters as identified in LR Section 1.1 for the 3,628 MWt NSSS power were used in the thermal analysis of the RCL and used in the evaluation for the pressurizer surge line. The RCL was reconciled for two temperature cases – one for the lower-bound temperature case, and the second for the upper-bound temperature case.

The RCL piping in the existing design basis was evaluated for the conditions associated with RCS with a hot leg temperature of 618.2°F, cross-over leg temperature of 557.8°F, and a cold leg temperature of 558.1°F. The reactor coolant upper bound temperatures for the SPU increase by 2.2°F for the hot leg, decrease by 0.2°F for the cross-over leg, and decrease by 0.1°F for the cold leg as compared to the current design basis temperatures. This temperature variation for the upper bound condition has an insignificant impact on the material properties – Young's modulus, coefficient of thermal expansion, support gaps, allowable stresses, and on the existing thermal analysis for the RCL piping. Considering RCL SPU lower bound temperature case, there is a temperature operating window as follows: 14.2°F between the upper bound T_{high} and lower bound T_{low} for the hot leg, 15.7°F between the upper bound T_{high} and lower bound T_{low} for the cross-over leg, and 15.8°F between the upper bound T_{high} and lower bound T_{low} for the cold leg. In the above-mentioned temperature operating window, the temperatures of the hot leg, cross-over leg, and cold leg for the RCL SPU lower bound temperature case are all lower than the corresponding temperatures of the hot leg, cross-over leg, and cold leg for the RCL SPU upper bound temperature case. The thermal piping stresses and displacements are dependent and directionally proportional to the coefficient of thermal expansion, and the coefficient of thermal expansion increases with increase in temperature. Therefore, the thermal piping loads and thermal stresses for the lower bound temperatures are lower than the corresponding loads and stresses for the upper bound case. Therefore, for the thermal case, the stresses for the upper bound case are higher, and the upper bound case piping stresses, primary equipment nozzle loads, primary equipment support loads, and the auxiliary line displacements at the connections to the RCL are limiting. Also, since the increase in the RCL SPU upper bound temperatures of the hot leg, cross-over leg, and cold leg are only 2.2°F or less in comparison to the hot leg, cross-over leg, and cold temperatures in the current RCL thermal analysis design basis, and this change is judged to have an insignificant impact on the current RCL thermal analysis design basis. Therefore, the impact on the RCL thermal analysis due to the SPU conditions is not significant in comparison to the current RCL design basis for the thermal case. So, for the upper temperature bounding case, the LBB loads at the RCL weld locations from the CPNPP Unit 2 as-built analysis are still applicable. However, the RCL piping thermal LBB loads for the lower bound case has been calculated based on the lower temperature bounding case and considered in the LBB evaluation.

The potential for stratification in the pressurizer surge line is increased as the difference in temperature between the pressurizer and the hot leg increases. Since the controlling ΔTs for the pressurizer surge line are associated with the plant heatup and cooldown events which are not affected by the uprate program, and the temperature changes of 10.8°F due to the SPU program has insignificant impact on the usage factor and ASME Code Section III equations 12 and 13 qualification. Therefore, the SPU has no adverse impact on either the thermal

stratification or the fatigue analysis for the pressurizer surge line, and the fatigue results for the surge line design basis stratification analysis as documented in Reference 4 remain valid.

The effect on the surge piping thermal loads at the weld locations for the temperature changes of 10.8°F due to the SPU program have been evaluated for the LBB loads (see LR subsection 2.1.6).

NSSS Design Transients (Unit 2)

The impact on design transients due to the changes in full-power operating temperatures for the power uprating program is addressed in LR subsection 2.2.6.

The impact due to the design transient changes for the CPNPP Unit 2 SPU on the RCL fatigue analysis was evaluated and found to be insignificant.

The controlling transients for the pressurizer surge line fatigue were evaluated and found to be not affected by the SPU program. Therefore, the SPU has no adverse effect on either the thermal stratification or the fatigue analysis for the pressurizer surge line documented in Reference 4.

The impact of changes to the design transients for the SPU on the RCL thermal loads and displacements at the intersection points of RCL piping centerline with the auxiliary piping nozzle centerline has been evaluated and found insignificant.

Loop LOCA Hydraulic Forcing Functions Forces and Associated Loop LOCA RPV Motions (Unit 2)

The impact on the RCL LOCA HFFs due to the SPU program is addressed in LR subsection 2.8.5.6.3.6 and the associated loop LOCA RPV motions are addressed in LR subsection 2.2.3. LBB is applicable for the RCL main loop piping and the pressurizer surge line (LR subsection 2.1.6), and also for the accumulator and residual heat removal (RHR) lines. The application of LBB criteria exempted these three large diameter RCS pipe breaks from consideration.

RCL LOCA analyses for the 4-inch pressurizer spray line break on the cold leg and the 6-inch safety injection line break on the hot leg with the loop LOCA hydraulic forcing function forces and the associated loop LOCA RPV motion for the SPU program were performed.

The resultant asymmetric external pressure loads on the RCP and steam generator resulting from a postulated pipe rupture and pressure buildup in the loop compartments are also considered in the RCL piping analysis for the SPU. Sub-compartment pressurization loads used in the CPNPP as-built analyses are assumed not to be affected by the power uprating, because the revised conditions do not require any changes to the sub-compartment forces which are used to calculate differential pressure loads.

Description of Analyses and Evaluations (Unit 2)

Since there are no changes in the weight of the system, deadweight analysis is not required for the SPU program. The results from the as-built seismic analysis for CPNPP Unit 2 (Reference 3) are not impacted either. Therefore, there are no changes to the reactor coolant loop deadweight and seismic analysis for the SPU program.

As previously described, the thermal analysis results from RCL as-built analysis are bounding and still applicable for CPNPP Unit 2 SPU.

The evaluation of the RCL includes performing RCL piping analyses with the revised LOCA HFFs and the associated RPV motions for the uprate program.

Based on the application of LBB, the analysis is performed for the breaks at the auxiliary nozzles for the 4-inch pressurizer spray line break on the cold leg, and 6-inch safety injection line break on the hot leg. Broken loop and unbroken loop time-history dynamic analyses are performed for these break cases.

The displacements obtained from the reactor vessel analysis (considering the effect of reactor vessel internals reaction and loop mechanical forces) are applied simultaneously with the loop hydraulic forces as imposed boundary conditions on the broken and unbroken loops to determine the deflection and stresses in the piping and loads on the supports.

The effect of the sub-compartment pressurization loads and the jet loads from the existing as-built analysis is also considered. The results (deflections and stresses in piping and loads on the supports) from these analyses are incorporated into results obtained from applications of corresponding loop hydraulic forces, reactor vessel motions and/or jet loads where applicable.

In addition, the impact on the RCL piping analyses due to the secondary side breaks due to the SPU on the existing piping analysis for the following secondary side breaks: a main steam line break at the steam generator nozzle, a main steam line break at the containment penetration anchor, a main feedwater line break at the steam generator nozzle, and an auxiliary feedwater line break at the steam generator nozzle, was evaluated and found insignificant.

The jet and thrust forces from the main steam line break, feedwater line break, and auxiliary line break calculated with the pressure for the SPU are bounded by the forcing functions, obtained with the pressure from the existing analysis. In addition, sub-compartment pressurization loads used in as-built RCL analyses are assumed not to be affected by the SPU, because the revised conditions do not require any changes to the sub-compartment forces that are used to calculate differential pressure loads. Therefore, the existing piping analysis results from the main steam nozzle break, feedwater break and auxiliary line break for the RCL as-built piping analysis are bounding and still applicable.

Additionally, as previously discussed, the current design basis pressurizer surge line analysis results including the effects of thermal stratification were reviewed and found still applicable for the SPU program.

The RCL piping, primary equipment nozzle and support displacements, displacements at the RCL branch nozzles from the deadweight, thermal, seismic, main steam nozzle break, feedwater nozzle break, and auxiliary feedwater line break as-built analyses as documented in Reference 3 remain valid for the SPU. However, the RCL LOCA piping, primary equipment nozzle and support displacements, displacements at the intersection points of RCL piping centerline with the auxiliary piping nozzle centerline from the RCL as-built LOCA have been changed due to the SPU program.

NSSS Piping, Components, and Supports Results (Unit 2)

Based on the evaluations performed for the SPU program NSSS PCWG design parameters, NSSS design transients, loop LOCA HFFs, and associated RPV motions, it is concluded that there is no adverse effect on the CPNPP Unit 2 current design basis RCL piping analyses and the current RCL piping design basis results in Reference 3 remain acceptable for the SPU program, except for the LOCA analyses results.

With the continued applicability of the design transients and the insignificant changes due to the thermal analysis, the impact on the fatigue evaluation of the RCL piping is negligible. Therefore, the existing design basis fatigue analysis of the reactor coolant loop piping (Reference 3) remains applicable for the SPU program.

Also, since there is no significant impact on the CPNPP Unit 2 RCL deadweight, thermal, seismic and secondary side pipe break line analyses due to the SPU program, the RCL deadweight, thermal, seismic and secondary side pipe break line displacements at the RCL branch nozzles (at the intersection points of RCL piping centerline with the auxiliary piping nozzle centerline) from the existing as-built analysis are still bounding and applicable for the uprate program.

The primary equipment nozzle loads and support loads from the existing as-built deadweight, seismic, thermal, and secondary side pipe break line analyses documented in Reference 3 are still bounding and applicable for the SPU program. Therefore, the existing design basis results from the deadweight, thermal, and seismic of the RCL piping analyses; the pressurizer surge line stratification analysis; primary equipment nozzle loads, and primary equipment support loads results from the deadweight, thermal, and seismic of the RCL piping analyses, remain applicable for the SPU program.

The RCL piping analysis for the breaks at the auxiliary nozzles for the 4-inch pressurizer spray line break and 6-inch safety injection line breaks for broken and unbroken loops were performed and results were shown to be acceptable.

The primary equipment nozzle loads RPV inlet and outlet nozzles, steam generator inlet and outlet nozzles, and RCP suction and discharge nozzles due the SPU have been evaluated with the new LOCA forces and RPV displacements for the uprate program and found to be acceptable.

The maximum RCL piping stresses for the RCL piping and the corresponding code-allowable stress values are presented in LR Table 2.2.2-2 for the SPU program and are the same as in the CPNPP Unit 2 as-built analysis, except for the Equation 9 Faulted condition stresses. The stresses are combined in accordance with the methods used in Reference 3 and as specified in the code criteria as described in Reference 1. From the results tabulated in the LR in Table 2.2.2-2, it can be concluded that the RCL piping stresses are within the allowable limits and meet the acceptance criteria (Reference 1) and are acceptable for the SPU program.

LBB pressure, deadweight, and seismic piping loads at the RCL weld locations from RCL piping as-built analysis remain bounding for the SPU program. The thermal LBB loads at the RCL weld locations at upper bounding thermal case from the existing CPNPP Unit 2 as-built analysis are still applicable for the SPU. The applicable RCL piping loads resulting from the range of operating temperatures, as defined by the NSSS parameters, were provided for evaluation and confirmation of LBB criteria (LR subsection 2.1.6).

The applicable thermal LBB loads, resulting from the temperature changes of 10.8°F, at the surge line weld locations due to the SPU program were provided for evaluation and confirmation of LBB (LR subsection 2.1.6). All other LBB piping loads (pressure, deadweight, and seismic loads) at the surge line welds from the existing design basis remain applicable to the SPU program.

NSSS Piping, Components, and Supports Conclusions (Unit 2)

The parameters associated with the SPU program have been evaluated for the following components:

- RCL piping stresses
- RCL piping system LBB loads for LBB evaluation
- RCL piping displacements at the RCL branch nozzles (at the intersection points of RCL piping centerline with the auxiliary piping nozzle centerline)
- Primary equipment nozzle loads
- Pressurizer surge line piping analysis including the effects of thermal stratification
- Primary equipment support loads (reactor vessel, steam generator, and RCP)

The evaluation indicated that the parameters associated with the SPU program have no adverse effect on the analysis of the RCL piping system, including impacts to the primary equipment nozzles. RCL piping stresses meet the required stress criteria as summarized in Table 2.2.2-2. The primary equipment nozzle loads are all acceptable. RCL piping loads transmitted for LBB evaluation for the SPU program are evaluated in LR subsection 2.1.6. The RCL primary equipment support loads are evaluated to meet the required stress criteria as

summarized in LR subsection 2.2.2.3 – Reactor Vessel and Supports, 2.2.2-5 – Steam Generator and Supports, and 2.2.2.2.6 – Reactor Coolant Pumps and Supports.

RCL piping displacements from the RCL deadweight thermal, seismic, and secondary side pipe breaks (main steam line break, feedwater line break and auxiliary feedwater line break) at branch nozzles (at the intersection points of RCL piping centerline with the auxiliary piping nozzle centerline) due to the SPU are not significantly impacted by the SPU program. Therefore, the existing deadweight, thermal, seismic and secondary side pipe break displacements from CPNPP Unit 2 are still applicable for the SPU program.

Additionally, the current design basis-analysis results for the pressurizer surge line, including the effects of thermal stratification, are still applicable and remain valid for the SPU program.

2.2.2.1.3 Conclusion

The evaluation related to the structural integrity of the RCL and its supports addressed the effect of the proposed SPU on these components and their supports. The evaluation has demonstrated that the RCL and their supports will continue to meet the requirements of 10 CFR 50.55a, GDC-1, -2, -4, -14, and -15 following implementation of the proposed SPU. Therefore, Luminant Power concludes that the proposed SPU is acceptable with respect to the structural integrity of the RCL and their supports.

2.2.2.1.4 References

1. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III, 1977 Edition and Addenda up to Summer 1979.

2. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III, Subsection NB, 1986 Edition.

3. WCAP-10197, Volume I, Revision 3, “Structural Analysis of the Reactor Coolant Loop for Comanche Peak Steam Electric Station Units 1 and 2, Analysis of the Primary Coolant Loop Piping.”

Table 2.2.2.1-1 Reactor Coolant Loop Stress Analysis Summary Applicable for the Power Uprating for CPNPP Unit 1						
Evaluation	**Hot Leg**		**Crossover Leg**		**Cold Leg**	
	Maximum	**Allowable(1)**	**Maximum**	**Allowable(1)**	**Maximum**	**Allowable(1)**
Eq. 9 Design Stress (ksi) (DW, P) Level A	14.67	28.35	17.57	28.35	17.54	28.35
Eq. 9 Design Stress (ksi) (DW, P, OBE) Level B	25.40	28.35	24.40	28.35	22.2	28.35
Eq. 9 Faulted Stress (ksi) (DW, P, SSE, Break Jet), Level D	53.50	56.70	30.70	56.7	26.1	56.7
Eq. 12 Stress (ksi)	31.33	56.70	6.51	56.70	20.17	56.70
Eq. 13 Stress (ksi)	56.80	57.4(2)	55.60	56.70	57.10	57.4(2)
Fatigue Usage Factor	0.85	1.0	0.12	1.0	0.22	1.0

Notes:

1. Allowable stress based on material type SA-351-CF8A at 650°F unless otherwise noted. See Note 2.
2. Allowable stress based on material type SA 351-CF8A at 620°F per paragraphs NB-3653 and NB-3222 of the Code (Reference 1).

Table 2.2.2.1-2 Reactor Coolant Loop Stress Analysis Summary Applicable for the Power Uprating for CPNPP Unit 2						
Evaluation	**Hot Leg**		**Crossover Leg**		**Cold Leg**	
	Maximum	**Allowable(1)**	**Maximum**	**Allowable(1)**	**Maximum**	**Allowable(1)**
Eq. 9 design stress (ksi) (DW, P) Level A	14.84	28.35	15.53	28.35	15.55	28.35
Eq. 9 design stress (ksi) (DW, P, OBE) Level B	20.70	28.35	22.68	28.35	22.31	28.35
Eq. 9 faulted stress (ksi) (DW, P, SSE, Break Jet), Level D	26.35	56.70	29.72	56.70	28.59	56.70
Eq. 12 stress (ksi)	31.33	56.70	6.51	56.70	20.17	56.70
Eq. 13 stress (ksi)	57.3	57.4(2)	55.60	56.70	57.10	57.4(2)
Fatigue usage factor	0.85	1.0	0.12	1.0	0.22	1.0

Notes:

1. Allowable stress based on material type SA-351-CF8A at 650°F unless otherwise noted. See Note 2.
2. Allowable stress based on material type SA-351-CF8A at 620.4°F per paragraphs NB-3653 and NB-3222 of the Code (Reference 1).

2.2.2.2 Balance-of-Plant Piping, Components, and Supports

2.2.2.2.1 Regulatory Evaluation

The regulatory evaluation included in Licensing Report (LR) subsection 2.2.2 also applies to balance-of-plant piping (BOP), components, and supports.

BOP piping, components, and supports for Comanche Peak Nuclear Power Plant (CPNPP) Units 1 and 2 are reviewed as part of the SPU. This section of the LR addresses piping, components and supports that are not included in LR subsection 2.2.2.1, NSSS Piping, Components, and Supports.

Current Licensing Basis

The generic current licensing basis in LR subsection 2.2.2 applies to BOP piping, components and supports, with the following specifics.

Final Safety Analysis Report (FSAR) sections that discuss the design of BOP piping, components, and supports include:

- FSAR Section 3.2, Classification of Structure, Components, and Systems, provides details with respect to the seismic classification of piping and piping components.

- FSAR Section 3.7B, Seismic Design, provides details with respect to the seismic qualification of piping and piping components.

- FSAR Section 3.9B, Mechanical Systems and Components, provides details with respect to the seismic qualification of piping and piping components.

2.2.2.2.2 Technical Evaluation

2.2.2.2.2.1 Introduction

BOP piping and support systems were evaluated to assess the impact of operating temperature, pressure and flow rate changes that will result due to the implementation of the SPU. The BOP piping and supports were evaluated to the following piping codes:

ASME Code Class 2 and 3 Piping

American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Division 1, Nuclear Power Plant Components, 1974 Edition including Summer 1974 Addenda Subsection NC and ND.

ASME Code Class 2 and 3 Pipe Supports

ASME Boiler and Pressure Vessel Code, Section III, Division 1, Nuclear Power Plant Components, 1974 Edition including the Winter 1974 Addenda Subsection NF.

In addition to the above codes, as permitted by ASME Section III, Subsection NA-1140 of the 1974 Edition of the Code, specific paragraphs in more recent editions and addenda of the ASME Code have been invoked.

Non-Nuclear Class (Class 5) Piping and Pipe Supports

ASME/American National Standards Institute (ANSI) B31.1 - 1973 Code for Pressure Piping through Winter 1974 Addenda.

The BOP piping and support systems that were evaluated for SPU conditions included the following systems:

- Main steam
- Feedwater (FW)
- Condensate

- Extraction steam
- Heater drains
- Steam generator blowdown
- Auxiliary feedwater
- Spent fuel pool cooling
- Residual heat removal
- Component cooling
- Service water
- Containment spray
- Chemical volume and control
- Safety injection
- Auxiliary steam

2.2.2.2.2.2 Description of Analyses and Evaluations

System operation at SPU conditions generally results in increased pipe stress levels and pipe support and equipment nozzle loads when those structures, systems, and components (SSCs) experience higher operating temperatures, pressures, or flow rates.

Current and SPU operating data (operating temperature, pressure and flow rate) were obtained from heat balance diagrams, calculations, and/or other reference documents.

Thermal, pressure and flow rate "change factors" were determined, as required, to compare and evaluate changes in SPU operating conditions. The change factors were based on the following ratios:

- The thermal change factor equals the ratio of the SPU to actual analyzed operating temperature. That is, thermal change factor is (T_{SPU}-70°F)/($T_{analyzed}$ 70°F).
- The pressure change factor was determined by the ratio of ($Pressure_{SPU}$/$Pressure_{analyzed}$).
- The flow rate "change factor" was determined by the ratio of (Flow $Rate_{SPU}$/ Flow $Rate_{analyzed}$)

Based on the magnitude of the calculated change factors, the following engineering activities were performed and/or conclusions reached.

For change factors less than or equal to 1.00 (that is, the current condition envelopes or equals the SPU condition), the piping and support system was concluded to be acceptable for SPU conditions.

For change factors greater than 1.00, an additional evaluation was performed to address the specific increase in temperature, pressure, and/or flow rate in order to determine piping and support system acceptability.

Flow rate increases due to the SPU occur mainly in systems related to the main power cycle. The two piping systems of most concern with respect to flow rate increases are the main steam and feedwater systems. The SPU flow rates and impact on potential flow-induced fluid transient loads were evaluated for the main steam and feedwater piping systems. The assessment of the main steam system identified that the existing flow rates considered in the current design basis fluid transient analyses used conservative (that is, bounding) values, that are higher than the main steam SPU flow rate. Hence, the main steam system was acceptable for SPU conditions and did not require any additional evaluations. However, an evaluation of the feedwater system was required to address the flow rate increase resulting from the SPU and its impact on fluid transient loads (that is, water hammer loads) resulting from feedwater isolation valve closure/check valve slam/feedwater pump trip events.

Changes in piping operating temperatures due to revised heat exchanger heat load requirements (such as component cooling and service water heat exchangers) have been addressed as part of the piping and support evaluations.

There were no changes to seismic inputs (amplified response spectra) or loads resulting from the SPU. The existing seismic design basis for all piping and supports remain valid and unaffected by the SPU. Hence, BOP piping and support seismic loadings will continue to meet the CPNPP current licensing basis with respect to the requirements of General Design Criterion (GDC)-2.

For BOP piping and support systems that required detailed analyses to reconcile SPU operating parameters, a summary of revised piping stress levels corresponding to SPU conditions are provided in Table 2.2.2.2-1 (Unit 1) and Table 2.2.2.2-2 (Unit 2). The results presented include current pipe stress levels, revised pipe stress levels for SPU conditions, allowable stress for the applicable loading condition, and the resulting design margin for each piping analysis that was evaluated to reconcile SPU conditions. The design margin provided is based on the ratio of the calculated stress divided by the allowable stress.

The following computer programs were used in performing the SPU piping and pipe support evaluations.

- NUPIPE-SWPC – The NUPIPE-SWPC program was used to perform detailed pipe stress analysis. This program is designed to perform analyses in accordance with the ASME Boiler and Pressure Vessel Code, Section III Nuclear Power Plant Components, and the ANSI/ASME B31.1 Power Piping Code.

- WATHAM-PC – The WATHAM-PC program was used to determine forcing functions for the feedwater fluid transient events. This program determines forcing functions in water-filled piping systems due to pump start/stop/trip, check valve closure, and valve opening and closing events. The forcing functions were input into a subsequent NUPIPE-SWPC pipe stress analysis run. The WATHAM-PC program is based on the method of characteristics numerical algorithm with a finite difference approximation in both space and time for the solution of one-dimensional homogenous, isothermal, and incompressible flow equations.

- PC-PREPS – PC-PREPS is a PC based computer program that performs a complete structural analysis, performing an American Institute of Steel Construction (AISC) code check, weld qualification, and baseplate/anchor bolt qualifications.

Other evaluations of issues that potentially impact BOP piping and supports are addressed in the following LR sections:

- Protection against dynamic effects, including GDC-4 requirements, of pipe whip and discharging fluids – LR subsection 2.2.1, Pipe Rupture Locations and Associated Dynamic Effects; and LR subsection 2.5.1.3, Pipe Failures.

- Protection against internally generated missiles and turbine missiles, including GDC-4 requirements, is discussed in LR subsection 2.5.1.2, Missile Protection.

- Design of the reactor coolant system and related components, including GDC-15 requirements, is discussed in LR subsection 2.2.2.1, NSSS Piping, Components, and Supports.

2.2.2.2.2.3 Results

Tables 2.2.2.2-1 and 2.2.2.2-2 provide a summary of current pipe stress levels, revised pipe stress levels for SPU conditions, and the resulting stress interaction ratios for each piping analysis that required detailed evaluation to reconcile SPU conditions. Piping systems not specifically listed in Tables 2.2.2.2-1 and 2.2.2.2-2 did not require detailed evaluation to reconcile SPU conditions or involve piping and support systems that will experience plant modifications. The stress results reported have incorporated thermal expansion and fluid transient increases, as applicable, that were reconciled as part of the SPU evaluations.

For piping systems that will experience plant modifications (see LR Section 1.0) to address SPU conditions, the piping and support evaluations will be performed as part of the overall design change package associated with the specific plant modification.

The piping and support evaluations performed concluded that all piping systems remain acceptable and will continue to satisfy design basis requirements when considering the temperature, pressure, and flow rate effects resulting from SPU conditions, with pipe support modifications if required in order to accommodate the revised support loads due to the SPU.

The results of the equipment nozzle and containment penetration evaluations concluded that these components remain within acceptable limits for SPU conditions.

In summary, the BOP piping, component, and support systems will continue to meet the CPNPP current licensing basis with respect to the requirements of GDCs-1, -2, -4, -14, and -15.

Additionally, the implementation of the SPU will result in higher flow rates for several piping systems. Piping systems experiencing these higher flow rates will be reviewed for potential vibration issues. Potentially affected piping will be included as part of the startup testing

program related to the overall implementation of the SPU. Refer to LR Section 2.12 for a discussion of the power ascension and testing plan.

2.2.2.2.3 Conclusion

Luminant Power concluded that the evaluations have adequately accounted for the effects of the proposed SPU on BOP piping, components, and supports. Based on this, it is concluded that the pressure-retaining components and their supports will continue to meet the CPNPP current licensing basis with respect to the requirements of 10 CFR 50.55a, GDC-1, -2, -4, -14, and -15. Luminant Power finds the proposed SPU acceptable with respect to the structural integrity of the pressure-retaining components and their supports.

Table 2.2.2.2-1

Stress Summary at SPU Conditions CPNPP Unit No. 1

Piping Analysis Description	Loading Condition	Current Stress (psi)	SPU Stress (psi)	Allowable Stress (psi)	Stress Interaction Ratio (Note 1)
FW Inside Containment (Penetration MI-0005 to Steam Generator TBX-RCPCSG-01)	Equation 9 (Upset)	13,270	13,876	18,000	0.771
	Equation 9 (Faulted)	25,962	25,962	36,000	0.721
FW Inside Containment (Penetration MI-0006 to Steam Generator TBX-RCPCSG-02)	Equation 9 (Upset)	11,652	12,349	18,000	0.686
	Equation 9 (Faulted)	42,435	42,435	48,000	0.884
FW Inside Containment (Penetration MI-0007 to Steam Generator TBX-RCPCSG-03)	Equation 9 (Upset)	13,915	14,818	18,000	0.823
	Equation 9 (Faulted)	34,526	34,526	36,000	0.959
FW Inside Containment (Penetration MI-0008 to Steam Generator TBX-RCPCSG-04)	Equation 9 (Upset)	17,036	18,560	24,000	0.773
	Equation 9 (Faulted)	34,833	34,833	36,000	0.968
FW Outside containment (Moment Restraint CP1-FWSSMR-04 to Penetration MI-0008)	Equation 9 (Upset)	17,402	17,453	18,000	0.97
	Equation 9 (Faulted)	34,960	34,248	36,000	0.95
FW Outside containment (Moment Restraint CP1-FWSSMR-02 to Penetration MI-0006)	Equation 9 (Upset)	17,558	17,575	18,000	0.98
	Equation 9 (Faulted)	35,245	30,737	36,000	0.85
FW Outside containment (Moment Restraint CP1-FWSSMR-01 to Penetration MI-0005)	Equation 9 (Upset)	15,002	17,850	18,000	0.99
	Equation 9 (Faulted)	32,261	34,151	36,000	0.95
FW Outside containment (Moment Restraint CP1-FWSSMR-03 to Penetration MI-0007)	Equation 9 (Upset)	17,130	16,414	18,000	0.91
	Equation 9 (Faulted)	34,331	32,372	36,000	0.90

Table 2.2.2.2-1 (cont.)

Stress Summary at SPU Conditions
CPNPP Unit No. 1

Piping Analysis Description	Loading Condition	Current Stress (psi)	SPU Stress (psi)	Allowable Stress (psi)	Stress Interaction Ratio (Note 1)
FW Outside Containment (SG Feedwater Pump CP1-FWAPFP-01 and CP1-FWAPFP-02 to Moment Restraints CP1-FWSSMR -01, -02, -03 and 04)	Equation 9 (Upset)	8,431	14,653	18,000	0.81
FW Outside Containment Main Bypass a/c 1-HV-2137	Equation 9 (Upset)	15,405	17,224	18,000	0.96
	Equation 9 (Faulted)	27,219	28,580	36,000	0.79
FW Outside Containment Main Bypass a/c 1-HV-2136	Equation 9 (Upset)	16,499	17,749	18,000	0.997
	Equation 9 (Faulted)	23,738	24,925	36,000	0.69
FW Outside Containment Main Bypass a/c 1-HV-2135	Equation 9 (Upset)	7,909	17,945	18,000	0.99
	Equation 9 (Faulted)	22,648	23,780	36,000	0.66
FW Outside Containment Main Bypass a/c 1-HV-2134	Equation 9 (Upset)	16,211	17,715	18,000	0.98
	Equation 9 (Faulted)	29,766	31,254	36,000	0.87
Condensate to FW Pumps Suction	Equation 14	27,974	31,890	37,500	0.85
Condenser to Condensate Pumps CP1-COAPCP-01, -02 Suction	Equation 13	737	811	22,500	0.04
Condensate Pumps CP1-COAPCP-01, -02 Discharge to Drain Coolers	Equation 13	10,130	11,548	22,500	0.51
Pressure Relief Valve from Condenser Dump Piping	Equation 14	24,951	28,358	37,500	0.76
Extraction Steam to Htrs 1-1A and 1B	Equation 13	17,887	18,066	22,500	0.8

Table 2.2.2.2-1 (cont.) Stress Summary at SPU Conditions CPNPP Unit No. 1					
Piping Analysis Description	**Loading Condition**	**Current Stress (psi)**	**SPU Stress (psi)**	**Allowable Stress (psi)**	**Stress Interaction Ratio (Note 1)**
Extraction Steam to Htrs 1-2A and 2B	Equation 13	15,883	16,042	22,500	0.72
Heater Drains Heater 1-1A, 1-2A to Main Condenser 1-A	Equation 13	16,533	16,698	22,500	0.74
Heater Drains Heater 1-4B to Heater 1-5B; Heater drain to Main Condenser 1-B	Equation 13	12,882	15,458	22,500	0.69
Heater Drains Heater 1-6B to Main Condenser 1-B	Equation 13	2,365	2,980	22,500	0.13
Heater Drains Shell Drain Tank 1-A to Condenser 1-B	Equation 14	34,413	35,038	37,500	0.94
Heater Drains Separator Drain Tank 1-A to Condenser 1-B	Equation 14	25,471	30,565	37,500	0.82
Heater Drains Shell Drain Tank 1-B to Condenser 1-B	Equation 14	23,634	28,361	37,500	0.76
Heater Drains Heater 1-1B and 1-2B to Main Condenser 1-B	Equation 13	17,430	20,916	22,500	0.93

Notes:

1. Stress Interaction Ratio reported is based on the ratio of SPU stress divided by the Allowable stress.

Table 2.2.2.2-2 Stress Summary at SPU Conditions CPNPP Unit No. 2					
Piping Analysis Description	**Loading Condition**	**Current Stress (psi)**	**SPU Stress (psi)**	**Allowable Stress (psi)**	**Stress Interaction Ratio (Note 1)**
FW Inside Containment (Penetration MI-0005 to Steam Generator TCX-RCPCSG-01)	Equation 9 (Upset)	14,053	14,350	18,000	0.797
	Equation 9 (Faulted)	26,655	27,063	36,000	0.752
FW Inside Containment (Penetration MI-0006 to Steam Generator TCX-RCPCSG-02)	Equation 9 (Upset)	17,242	17,693	18,000	0.983
	Equation 9 (Faulted)	23,768	24,110	36,000	0.670
FW Inside Containment (Penetration MI-0007 to Steam Generator TCX-RCPCSG-03)	Equation 9 (Upset)	14,668	14,989	18,000	0.833
	Equation 9 (Faulted)	24,879	25,245	36,000	0.701
FW Inside Containment (Penetration MI-0008 to Steam Generator TCX-RCPCSG-04)	Equation 9 (Upset)	15,869	16,204	18,000	0.900
	Equation 9 (Faulted)	23,864	25,187	36,000	0.672
FW Outside containment (Moment Restraint CP2-FWSSMR-04 to Penetration M2-0008)	Equation 9 (Upset)	11,400	11,970	18,000	0.67
	Equation 9 (Faulted)	31,518	33,094	36,000	0.92
FW Outside containment (Moment Restraint CP2-FWSSMR-02 to Penetration M2-0006)	Equation 9 (Upset)	9,222	9,683	18,000	0.54
	Equation 9 (Faulted)	32,307	33,922	36,000	0.94
FW Outside containment (Moment Restraint CP2-FWSSMR-01 to Penetration M2-0005)	Equation 9 (Upset)	9,381	9,850	18,000	0.55
	Equation 9 (Faulted)	34,514	35,931	36,000	0.998
FW Outside containment (Moment Restraint CP2-FWSSMR-03 to Penetration M2-0007)	Equation 9 (Upset)	10,153	10,661	18,000	0.59
	Equation 9 (Faulted)	31,798	33,388	36,000	0.93

Table 2.2.2.2-2 (cont.) Stress Summary at SPU Conditions CPNPP Unit No. 2					
Piping Analysis Description	**Loading Condition**	**Current Stress (psi)**	**SPU Stress (psi)**	**Allowable Stress (psi)**	**Stress Interaction Ratio (Note 1)**
FW Outside Containment (SG Feedwater Pump CP2-FWAPFP-01 and CP2-FWAPFP-02 to Moment Restraints CP2-FWSSMR -01, -02, -03 and 04)	Equation 9 (Upset)	15,628	16,409	18,000	0.91
	Equation 9 (Faulted)	15,628	16,409	36,000	0.46
FW Outside Containment Main Bypass a/c2-HV-2137	Equation 9 (Upset)	16,261	17,074	18,000	0.95
	Equation 9 (Faulted)	27,726	29,112	36,000	0.81
FW Outside Containment Main Bypass a/c 2-HV-2136	Equation 9 (Upset)	13,428	14,099	18,000	0.78
	Equation 9 (Faulted)	17,854	18,747	36,000	0.52
FW Outside Containment Main Bypass a/c 2-HV-2135	Equation 9 (Upset)	14,737	15,474	18,000	0.86
	Equation 9 (Faulted)	31,276	32,840	36,000	0.91
FW Outside Containment Main Bypass a/c 2-HV-2134	Equation 9 (Upset)	12,940	13,587	18,000	0.75
	Equation 9 (Faulted)	30,666	32,199	36,000	0.89
FW Outside Containment Bypass to Condenser	Equation 9 (Upset)	14,181	14,890	18,000	0.83
Aux FW Inside Cont	Equation 9 (Upset)	11,879	12,133	18,000	0.674
FW-2-152	Equation 9 (Faulted)	35,242	35,868	36,000	0.996
Aux FW Inside Cont	Equation 9 (Upset)	12,704	12,997	18,000	0.722
FW-2-153	Equation 9 (Faulted)	35,054	35,674	36,000	0.991
Aux FW Inside Cont	Equation 9 (Upset)	10,598	10,811	18,000	0.601
FW-2-154	Equation 9 (Faulted)	32,892	33,476	36,000	0.930

Table 2.2.2.2-2 (cont.)

Stress Summary at SPU Conditions
CPNPP Unit No. 2

Piping Analysis Description	Loading Condition	Current Stress (psi)	SPU Stress (psi)	Allowable Stress (psi)	Stress Interaction Ratio (Note 1)
Aux FW Inside Cont	Equation 9 (Upset)	8,863	8,989	18,000	0.499
FW-2-155	Equation 9 (Faulted)	35,310	35,947	36,000	0.999
Condensate to Condensate Pumps CP2-COAPCP-01, -02 Suction	Equation 13	844	895	22,500	0.04
Extraction Steam to Htrs 3A and 3B	Equation 14	3,719	3,756	22,500	0.17
Extraction Steam to Htrs 4A and 4B	Equation 13	2,098	2,119	22,500	0.09
Heater Drains Htrs 2B to Condenser	Equation 13	17,430	20,567	22,500	0.91
Heater Drains Moisture Separator Reheater A Drains to Seperator Drain Tank 1A	Equation 14	25,471	30,056	37,500	0.80

Notes:

1. Stress Interaction Ratio reported is based on the ratio of SPU stress divided by the Allowable stress.

2.2.2.3 Reactor Vessel and Supports

2.2.2.3.1 Regulatory Evaluation

The reactor pressure vessel (RPV) and its supports are reviewed as part of the SPU described in Final Safety Analysis Report (FSAR) Section 5.3. The RPV supports are described FSAR subsection 5.4.14.2.1. The Regulatory Evaluation included in Licensing Report (LR) subsection 2.2.2 also applies to the RPV and its supports.

Current Licensing Basis

The generic current licensing basis in LR subsection 2.2.2 applies to the RPV and its supports, with the following amplifications.

The Comanche Peak Nuclear Power Plant (CPNPP) RPV is cylindrical, with a welded hemispherical bottom head and a removable, flanged and gasketed, hemispherical upper head. The vessel contains the core, core support structures, control rods, and other components directly associated with the core.

The vessel has inlet and outlet nozzles located in a horizontal plane below the RPV flange but above the top of the core. Coolant enters the vessel through the inlet nozzles, flows down the core barrel-vessel wall annulus, and is then redirected at the bottom to flow up through the core and out the outlet nozzles.

FSAR Section 5.3.1 states, in part, that all pressure boundary materials used in the RPV are selected and fabricated in accordance with the requirements of Section III of the American Society of Mechanical Engineers (ASME) Code. Table 5.2-1 of the FSAR provides ASME Boiler and Pressure Vessel (B&PV) Code Edition and Addenda applicable to the RPV. A general discussion of materials specifications is given in FSAR Section 5.2.3, with types of materials listed in FSAR Table 5.2-2.

Section 5.3.1 of the FSAR states, in part, that:

- The RPV is Safety Class 1. Design and fabrication of the RPV was carried out in strict accordance with the ASME Code, Section III, Class 1 requirements.
- The head flange for Unit 2 and nozzles for Units 1 and 2 were manufactured as forgings. The cylindrical portion of the vessel is made up of several shells, each consisting of formed plates joined by full-penetration longitudinal weld seams. The hemispherical head for Unit 2 was made with a dished plate. The integral parts of the vessel and closure head subassemblies were joined by welding, primarily using the single or multiple wire submerged arc. A replacement reactor vessel closure head was recently installed at CPNPP Unit 1 during the spring 2007 outage. The replacement head was designed and fabricated as a one-piece forging in accordance with the ASME Code Section III, 1989 Edition with no addenda.

The RPV and RPV supports are designed to withstand stresses originating from various operating design transients described in FSAR Section 3.9N.1.1. The RPV supports are designed to meet the same Safety Class designation as the components they support. The RPV supports are classified Seismic Category I, as stated in FSAR Table 17A-1.

FSAR Section 5.4.14.2.1 states, in part, that:

Supports for the reactor vessel are individual, air-cooled rectangular box structures beneath the vessel nozzles bolted to the primary shield wall concrete. Each box structure consists of a horizontal top plate that receives loads from the reactor vessel shoe, a horizontal bottom plate supported by short columns that transfer the loads to the primary shield wall concrete, and connecting vertical plates. The supports are air cooled to maintain the supporting concrete temperature within acceptable levels.

FSAR Table 3.9N-1 summarizes RCS design transients. It states that the RPV is designed for 200 heatup transients of 100°F per hour, and an additional 200 cooldown transients of 100°F per hour.

FSAR Table 5.3-5 provides the RPV design data. The ability of the pressure boundary components to perform throughout the design lifetime as defined in the design specification is confirmed by the stress analysis report required by the ASME Code, Section III.

2.2.2.3.2 Technical Evaluation

2.2.2.3.2.1 Introduction

To ensure the adequacy of the CPNPP Units 1 and 2 reactor vessels and their supports, evaluations were performed for the revised operating conditions of the SPU, including temperature and transient effects. The evaluations were based on the design basis calculations that previously qualified the components and, if appropriate, the values from the design basis were updated.

2.2.2.3.2.2 Input Parameters, Assumptions, and Acceptance Criteria

Input Parameters

The analyses and evaluations that were performed for the vessel components and supports use the design basis calculations and results in combination with the revised operating temperatures and RCS transients associated with the CPNPP Units 1 and 2 SPU. The design basis calculations, which are contained in the appropriate design/stress reports and their addenda, were based on the current operating conditions for each unit, including temperature and transient effects, and seismic and LOCA loads.

Assumptions

There were no major assumptions associated with the CPNPP Units 1 and 2 reactor vessel evaluations.

Acceptance Criteria

Analyses and evaluations were required in accordance with Section III of the ASME B&PV Code to substantiate the structural adequacy of the CPNPP Units 1 and 2 reactor vessels for operation under uprate conditions. Revised maximum stress intensity ranges and cumulative fatigue usage factors were calculated and compared to the following acceptance criteria:

1. The maximum range of primary-plus-secondary stress intensity resulting from mechanical and thermal loads shall not exceed $3S_m$ at operating temperature. In addition to the above, in the event the primary-plus-secondary stress intensity resulting from mechanical and thermal loads exceeds the $3S_m$ acceptance criteria, the design shall be considered acceptable if the criteria specified for a simplified elastic-plastic analysis per Section NB-3228.3 of the ASME B&PV Code, Section III, Division 1 1971 Edition through Winter 1972 Addenda (for the Unit 1 vessel not including the replacement closure head and the Unit 2 vessel) and the 1989 Edition with no addenda can be met.

2. The maximum cumulative usage factor resulting from the peak stress intensities due to the normal and upset condition design transient mechanical and thermal loads cannot exceed 1.0 in accordance with the procedure outlined in the ASME B&PV Code, Section III, Division 1, 1971 Edition through Winter 1972 Addenda (for the Unit 1 vessel not including the replacement closure head, and the Unit 2 vessel) and the 1989 Edition with no addenda (for the Unit 1 replacement closure head and attachments).

2.2.2.3.2.3 Description of Analyses and Evaluations

The stress intensities and fatigue usage factors of the reactor vessel, including the closure head and main closure region components, were updated based on the uprate operating conditions.

For Unit 1, the current RCS design transients (LR subsection 2.2.6) and operating temperatures (LR subsection 1.1) were determined to be applicable for the SPU. Therefore, the current vessel evaluations and the results of the design analyses performed for the Replacement Head Program are applicable for the uprate for Unit 1.

For Unit 2, the SPU program identified many of the transients as being applicable to the Unit 2 uprate. Some primary side transients had to be revised, however, due to differences in the steam generator models. As a result, the evaluation of the Unit 2 vessel considered a combination of revised uprate and current Unit 1 transients. No changes to the pressure transients were identified for the SPU program. Therefore, only thermal transient revisions had to be considered in the vessel evaluations.

A transient comparison was first performed in which the revised RCS design transients for the CPNPP SPU were compared to those for the existing qualifications, and where found to be more severe than the existing qualifications, were considered in the vessel evaluation for the uprate. The operating temperature ranges for the uprate were also compared to those for the existing qualifications.

The stress intensities for those transients that were deemed more severe than their baseline counterparts were examined to determine their effect on the maximum ranges of stress intensity for all the regions of the reactor vessel. The changes in the thermal stresses, due to variations from the baseline transients, were evaluated using standard engineering approaches. The incremental thermal stress changes were then factored into stress intensities that are documented in the baseline stress report(s) and the effects of the changes on the maximum ranges of stress intensity were observed and documented.

The peak stress intensity ranges for the fatigue evaluation were also adjusted to account for the incremental thermal stress changes caused by adverse changes to the baseline transients. The peak thermal stresses were multiplied by the appropriate scaling factor, where necessary, before determining a new peak stress intensity range and finally an alternating stress. The allowable number of cycles of alternating stress was found from the applicable fatigue curve in the ASME B&PV Code, Section III, Division 1, 1971 Edition through Winter 1972 Addenda for the Unit 1 vessel (not including the closure head) and the Unit 2 vessel including the closure head, and the cumulative fatigue usage factors were revised accordingly.

Where applicable, the maximum and minimum stress intensity ranges and fatigue usage factors were revised to reflect the presented changes to the baseline transients. In other cases, the baseline stress analysis in the baseline stress report remained conservative with regard to the design transients and new calculations were not necessary. For those cases, the maximum stress intensity ranges and fatigue usage factors reported in the baseline reactor vessel stress report were not changed.

For both units, evaluations for seismic and LOCA loads were not performed because the previously evaluated seismic and LOCA loads were determined to be applicable to the uprate.

The reactor vessel supports were evaluated and found acceptable for the revised operating conditions resulting from the SPU, including temperature and transient effects. The existing design basis calculations that performed the qualification and demonstrated the acceptability of the reactor vessel support loads were reviewed to determine if the previously qualified loads envelope the revised loads.

2.2.2.3.2.4 Results

The stress intensity ranges and fatigue usage factors for the Unit 1 vessel are unchanged from those calculated in the current vessel evaluations. The stress intensity ranges and usage factors for the Unit 2 vessel are, in some cases, slightly higher than those presented for Unit 1 due to additional transient revisions that apply to Unit 2 for the uprate. Based on the reactor vessel evaluations outlined in this report, all of the maximum ranges of primary-plus-secondary stress intensity and maximum cumulative fatigue usage factors for the CPNPP Units 1 and 2 reactor vessel components listed in Tables 2.2.2.3-1 and 2.2.2.3-2 continue to satisfy the applicable limits of ASME B&PV Code, Section III, Division 1, 1971 Edition through Winter 1972 Addenda (for the Unit 1 vessel not including the replacement closure head, and the Unit 2 vessel) and the 1989 Edition with no addenda (for the Unit 1 replacement closure head and attachments).

LOCA hydraulic forces are unchanged from the previously evaluated forces. In addition, the SPU has no impact on seismic response. Since the current reactor vessel evaluation indicated acceptability of all faulted loads, no additional load evaluation was required for the SPU program.

The CRDM housing moments are shown and evaluated in Table 2.2.2.3-3. Loads due to the SPU are less than the allowable or limiting loads.

The reactor vessel supports were evaluated for the SPU conditions and found to be acceptable.

2.2.2.3.3 Conclusions

Luminant Power has reviewed the evaluations related to the structural integrity of the reactor vessel and vessel supports and concludes that the evaluations have adequately addressed the effects of the proposed SPU on the reactor vessel and vessel supports. Luminant Power further concludes that the evaluations have demonstrated that the reactor vessel and vessel supports continue to meet the requirements of 10 CFR 50.55a, General Design Criterion (GDC)-1, -2, -4, -14, and -15 following implementation of the proposed SPU. Therefore, Luminant Power finds the proposed SPU acceptable with respect to the design of the reactor vessel and vessel supports for Units 1 and 2.

Table 2.2.2.3-1

Maximum Range of Stress Intensity and Cumulative Fatigue Usage Factors for CPNPP Unit 1

Location	Maximum Range of Stress intensity	Cumulative Fatigue Usage Factor
CRDM Housings	63.4 ksi < $3S_m$ = 69.9 ksi	0.174
Closure Head/Flange	64.0 ksi < $3S_m$ = 80.1 ksi	0.081
Vessel Flange	50.3 ksi < $3S_m$ = 80.1 ksi	0.045
Closure Studs	94.9 ksi < $2.7S_m$ = 99.6 ksi	0.594
Vessel Wall Transition	56.3 ksi < $3S_m$ = 80.1 ksi	0.022
Outlet Nozzles and Support Pads	Outlet Nozzle Safe End 44.4 ksi < $3S_m$ = 52.3 ksi Outlet Nozzle 75.4 ksi < $3S_m$ = 80.1 ksi Support Pad 46.4 ksi < $3S_m$ = 80.1 ksi	Outlet Nozzle 0.328 Support Pad 0.025
Inlet Nozzles and Support Pads	Inlet Nozzle Safe End 42.0 ksi < $3S_m$ = 52.3 ksi Inlet Nozzle 66.7 ksi < $3S_m$ = 80.1 ksi Support Pad 61.4 ksi < $3S_m$ = 80.1 ksi	Inlet Nozzle 0.112 Support Pad 0.030
Core Support Pads (Lower Radial Keys)	43.2 ksi < $3S_m$ = 80.1 ksi (Wall) 43.2 ksi < $3S_m$ = 69.9 ksi (Pad)	0.117
Bottom Head to Shell Juncture	41.8 ksi < $3S_m$ = 80.1 ksi	0.012
Bottom-Mounted Instrumentation (BMI) Tubes	70.1[1] ksi > $3S_m$ = 69.9 ksi	0.409

Note:

1. The maximum range of stress intensity for the BMI is justified by using simplified elastic-plastic analysis methods per Subsection NB-3228.3 of the ASME B&PV Code.

Table 2.2.2.3-2 Maximum Range of Stress Intensity and Cumulative Fatigue Usage Factors for CPNPP Unit 2		
Location	**Maximum Range of Stress intensity**	**Cumulative Fatigue Usage Factor**
CRDM Housings	59.0 ksi < $3S_m$ = 69.9 ksi	0.192
Closure Head/Flange	55.7 ksi < $3S_m$ = 80.1 ksi	0.041
Vessel Flange	50.3 ksi < $3S_m$ = 80.1 ksi	0.045
Closure Studs	94.9 ksi < $2.7S_m$ = 99.6 ksi	0.594
Vessel Wall Transition	56.3 ksi < $3S_m$ = 80.1 ksi	0.022
Outlet Nozzles and Support Pads	Outlet Nozzle Safe End 44.4 ksi < $3S_m$ = 52.3 ksi Outlet Nozzle 75.4 ksi < $3S_m$ = 80.1 ksi Support Pad 46.4 ksi < $3S_m$ = 80.1 ksi	Outlet Nozzle 0.369 Support Pad 0.027
Inlet Nozzles and Support Pads	Inlet Nozzle Safe End 42.0 ksi < $3S_m$ = 52.3 ksi Inlet Nozzle 66.7 ksi < $3S_m$ = 80.1 ksi Support Pad 61.4 ksi < $3S_m$ = 80.1 ksi	Inlet Nozzle 0.115 Support Pad 0.032
Core Support Pads (Lower Radial Keys)	43.2 ksi < $3S_m$ = 80.1 ksi (Wall) 43.2 ksi < $3S_m$ = 69.9 ksi (Pad)	0.117
Bottom Head to Shell Juncture	41.8 ksi < $3S_m$ = 80.1 ksi	0.012
BMI Tubes	70.1[(1)] ksi > $3S_m$ = 69.9 ksi	0.489
Head Adapter Plugs	41.2 ksi < $3S_m$ = 48.6 ksi	0.021

Note:

1. The maximum range of stress intensity for the BMI is justified by using simplified elastic-plastic analysis methods per Subsection NB-3228.3 of the ASME B&PV Code.

Table 2.2.2.3-3 Reactor Vessel CRDM Housings Applied Moments		
Location	**SRSS (SSE+LOCA) in-lb**	**Limit in-lb**
Head Adapter (stainless steel)	136,320	212,000
Head Adapter (Inconel)	200,310	240,000

2.2.2.4 Control Rod Drive Mechanism

2.2.2.4.1 Regulatory Evaluation

The evaluation of the Comanche Peak Nuclear Power Plant (CPNPP) Units 1 and 2 control rod drive mechanisms (CRDMs) presented herein is an assessment of the impact on the structural integrity of the assemblies from the thermal transients and maximum operating temperatures and pressures that result from the proposed SPU operating conditions. The results of the structural analysis of the CRDM pressure boundary show that the analyzed stresses do not exceed the stress allowable of the American Society of Mechanical Engineers (ASME) Code, and that the cumulative fatigue usage factors from the code fatigue analysis remain less than 1.0.

The CRDMs are mounted onto the reactor vessel (RV) head by means of head adaptors welded to RV head penetrations. The CRDM assembly consists of the internal (latch) assembly, the pressure vessel, the operating coil stack, the drive shaft assembly, and the position indicator coil stack. Main coolant fills the pressure containing parts of the drive mechanism. Therefore, the pressure vessel component of the CRDM assembly constitutes a reactor coolant pressure boundary. The CRDMs and all the components of the control rod drive system are designed as Seismic Category I equipment.

This section addresses the ASME Code of record structural considerations for the pressure boundary components of the full-length L-106 CRDMs. The CRDMs were evaluated using the NSSS operating parameters of Licensing Report (LR) Section 1.1, Nuclear Steam Supply System Parameters, and the NSSS design transients subsection 2.2.6, NSSS Design Transients developed, for the CPNPP Units 1 and 2 SPU Program.

Current Licensing Basis

The current licensing basis contained in LR subsection 2.2.2 applies to this section.

A description of the CRDMs is contained in Final Safety Analysis Report (FSAR) Section 3.9N.4. FSAR Section 4.2.2.3 (In-Core Components) contains a description of CRDMs with respect to reactivity control. FSAR Table 5.2-2 shows the CRDM materials.

2.2.2.4.2 Technical Evaluation

Input Parameters, Assumptions, and Acceptance Criteria

The Model L-106C CRDM in Unit 1 was originally designed and analyzed to meet the requirements of the ASME Boiler and Pressure Vessel (B&PV) Code, Section III, Subsection NB and Appendices, 1989 Edition, no Addenda. The 1974 Edition through Summer 1974 Addenda was the code of record for the original design and analysis of the Model L-106A CRDM in Unit 2. These original analyses were used as input for this evaluation. As explained above, the NSSS operating parameters and NSSS design transients developed for the CPNPP Units 1 and 2 SPU

are used as the new inputs for this evaluation. The seismic loading has not been changed for the CPNPP Units 1 and 2 SPU Program.

The CPNPP Units 1 and 2 CRDMs are the cold head type, defined by the vessel inlet reactor coolant temperature, T_{cold}, in LR Section 1.1, Nuclear Steam Supply System Parameters. For conservatism, the most bounding transient, either T_{cold} or T_{hot}, was used for the evaluation of the CRDMs.

Acceptance Criteria

The acceptance criteria for the ASME Code structural analysis of the CRDM reactor coolant pressure boundary are that the analyzed stresses do not exceed the stress allowable of the ASME Code, and that the cumulative fatigue usage factors from the code fatigue analysis remain less than 1.0. For those cases for which changes to the design transients would have allowed a decrease in stresses or cumulative usage factors, no decrease was calculated, and no credit was taken for such a decrease.

Description of Analyses and Evaluations

Operating Pressure and Temperature

The NSSS temperatures and pressures developed for the CPNPP Units 1 and 2 SPU Program as given in LR Section 1.1, Nuclear Steam Supply System Parameters, were compared to those used for the current design analyses for the CRDMs. There is no change in the reactor coolant pressure of 2,250 psia for any of the SPU cases. The hot leg temperature (T_{hot}) defined by the vessel outlet temperature is a maximum of 620.4°F, which is less than the 650.0°F temperature used in the current analysis of record. Since none of the temperatures exceed the previously analyzed temperatures, and the pressure does not change, the NSSS parameters developed for the SPU program and used for this evaluation are bounded by the current analyses of record.

Transient Discussion

The NSSS design transients, discussed in LR subsection 2.2.6, NSSS Design Transients, were not significantly changed from the currently analyzed condition for the CPNPP Units 1 and 2 CRDMs. For Unit 1, there were no changes to the design transients. Therefore, the current transients evaluated are acceptable for the SPU program. For Unit 2, the changes consisted of the following:

- 33 transients not specified in the existing design basis L-106A CRDM analysis
- Temperature and pressure range difference between the SPU and existing design basis NSSS design transients

The 33 additional transients incorporated for the uprate evaluation are not considered significant because they are bounded by the most severe transients used in the analysis of record. The

impact these changes had on the stress intensities were analyzed using the generic stress report methodology and are acceptable for the CPNPP SPU.

Results

A summary of the stress results of the evaluations performed for the SPU is presented in Tables 2.2.2.4-1 and 2.2.2.4-4 through 2.2.2.4-6. The changes in calculated stress intensities were proven acceptable for the CPNPP Units 1 and 2 SPU operating conditions. The calculated stresses in all but three pressure vessel parts of the CRDMs meet the allowable ASME stress limits. These three cases were determined to be acceptable based on the following:

- The maximum primary-plus-secondary-stress-intensity ranges for Unit 1 top cap and lower section exceed the allowable $3S_m$ when including the design condition seismic load factors. It was concluded that these stress intensities satisfy the criteria for upset condition (including seismic stresses) by meeting the requirements of section NB-3228.5(b) and NB-3222.5 of the ASME Code.

- The maximum lower joint canopy primary-plus-secondary-stress intensity of 45,985 psi exceeded the allowable by .2 percent. This is considered insignificant due to the conservatism that the allowable is based on the design temperature of 650°F as opposed to the actual nodal temperature or 78°F. The ASME Code allowable stress intensity $3S_m$ is 60 ksi at 78°F and 45.9 ksi at 650°F.

The cumulative usage factors that were calculated are given in Tables 2.2.2.4-2 and 2.2.2.4-3. The cumulative usage factors for Unit 1 do not change due to the SPU program. The cumulative usage factors for Unit 2, given in Table 2.2.2.4-3, remain bounded for the SPU program. The highest cumulative usage factor for Unit 2, (0.858), was calculated at the upper joint canopy. The usage factor was calculated in a conservative manner. The applied transients were grouped and the allowable number of cycles considered for each group was based on the most severe transient in the group.

2.2.2.4.3 Conclusions

Luminant Power has reviewed the evaluation related to the structural integrity of pressure-retaining components of the CRDM. For the reasons presented above, Luminant Power concludes that the effects of the proposed SPU on these components have been adequately addressed. It is further concluded that these pressure-retaining components will meet the requirements of 10 CFR 50.55a, General Design Criterion (GDC)-1, -2, -4, -14, and -15 following implementation of the proposed SPU. Therefore, Luminant Power finds the proposed SPU acceptable with respect to the structural integrity of the pressure-retaining components.

Table 2.2.2.4-1

Unit 1 Stress Summary

		Design Condition		Normal/Upset Condition		Testing Condition		Special Condition	
Component	**Param. Per ASME Code III**	**Calc (psi)**	**Allow (psi)**	**Calc (psi)**	**Allow (psi)**	**Calc (psi)**	**Allow (psi)**	**Calc (psi)**	**Allow (psi)**
Top Cap	P_m	4,000	16,200	N/A	N/A	6,000	27,000	N/A	N/A
	P_m+P_b	5,900	24,300	N/A	N/A	8,880	40,500	N/A	N/A
	P_m+P_b+Q	N/A	N/A	31,270	48,600	N/A	N/A	N/A	N/A
		N/A	N/A	64,390[1]	48,600	N/A	N/A	N/A	N/A
	Thermal Stress Ratchet	N/A	N/A	25,410	97,810	N/A	N/A	N/A	N/A
	$\sigma_1+\sigma_2+\sigma_3$	N/A	N/A	N/A	N/A	N/A	N/A	19,240	80,000
Lower Section	P_m	13,970	16,200	N/A	N/A	20,960	27,000	N/A	N/A
	P_m+P_b	16,160	24,300	N/A	N/A	24,230	39,350	N/A	N/A
	P_m+P_b+Q	N/A	N/A	48,780	51,810	N/A	N/A	N/A	N/A
		N/A	N/A	81,050[1]	51,810	N/A	N/A	N/A	N/A
	Thermal Stress Ratchet	N/A	N/A	35,590	42,300	N/A	N/A	N/A	N/A
	Bearing	N/A	N/A	N/A	N/A	N/A	N/A	3,433	17,900
	$\sigma_1+\sigma_2+\sigma_3$	N/A	N/A	N/A	N/A	N/A	N/A	38,180	80,000

Note:

1. Includes design condition seismic load factors.

Table 2.2.2.4-2 Unit 1 Cumulative Fatigue Usage Factors			
Component		**Total Usage Factor**	**Allowable Usage Factor**
Top Cap	w/o seismic	0.0431	1.0
	w/seismic	0.2838	1.0
Lower Section	w/o seismic	0.4938	1.0
	w/seismic	0.9719	1.0

Table 2.2.2.4-3 Unit 2 Cumulative Fatigue Usage Factors			
Joint	**Component**	**Total Usage Factor**	**Allowable Usage Factor**
Upper	Cap	0.0	1.0
	Road travel housing	0.0	1.0
	Canopy	0.858	1.0
	Weld canopy	0.549	1.0
	Threaded area	0.360	1.0
Middle	Road travel housing	0.0	1.0
	Latch housing	0.0	1.0
	Canopy	0.0	1.0
	Weld canopy	0.524	1.0
	Threaded area	0.034	1.0
Lower	Latch housing	0.0	1.0
	Head adaptor	0.0	1.0
	Canopy	0.010	1.0
	Weld canopy	0.0242	1.0
	Threaded area	0.028	1.0

Table 2.2.2.4-4
Unit 2 Upper Joint Stress Summary

Upper Joint		Design Condition		Normal Condition		Upset Condition		Testing Condition		Special Condition		Faulted Condition	
Component	Param. Per ASME Code III	Calc (psi)	Allow (psi)	Calc (psi)	Allow (psi)	Calc (psi)	Allow (psi)	Calc (psi)	Allow (psi)	Calc (psi)	Allow (psi)	Calc (psi)	Allow (psi)
Cap	P_m	5,954	16,100	N/A	N/A	N/A	N/A	7,400	16,110	N/A	N/A	7,216	38,640
	P_m+P_b	20,757	24,150	N/A	N/A	N/A	N/A	22,203	24,165	N/A	N/A	22,019	57,960
	P_m+P_b+Q	N/A	N/A	19,107	48,300	19,113	48,300	N/A	N/A	N/A	N/A	N/A	N/A
	$\sigma_1+\sigma_2+\sigma_3$	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	-16,522	64,400	N/A	N/A
Rod Travel Housing	P_m	14,172	16,100	N/A	N/A	N/A	N/A	17,613	21,420	N/A	N/A	17,176	38,640
	P_m+P_b	19,419	24,150	N/A	N/A	N/A	N/A	20,826	32,130	N/A	N/A	20,389	57,960
	P_m+P_b+Q	N/A	N/A	23,574	48,300	21,106	48,300	N/A	N/A	N/A	N/A	N/A	N/A
	$\sigma_1+\sigma_2+\sigma_3$	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	13,922	64,400	N/A	N/A
Canopy	P_m	4,606	16,100	N/A	N/A	N/A	N/A	5,724	16,110	N/A	N/A	5,582	38,640
	P_m+P_b	8,254	24,150	N/A	N/A	N/A	N/A	9,372	24,165	N/A	N/A	9,230	57,960
	P_m+P_b+Q	N/A	N/A	27,594	48,300	40,057	48,300	N/A	N/A	N/A	N/A	N/A	N/A
	$\sigma_1+\sigma_2+\sigma_3$	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	9,667	64,400	N/A	N/A
Threaded Area	Pm (Shear)	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	5,370	9,660	N/A	N/A
	2x Shear	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	38,020	48,300	N/A	N/A
	P_m+P_b+Q	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	47,500	48,300	N/A	N/A
	Bell Mouthing Stress Intensity	N/A	N/A	19,695	20,487	20,874	21,611	N/A	N/A	N/A	N/A	N/A	N/A

Table 2.2.2.4-5 Unit 2 Middle Joint Stress Summary													
Middle Joint		Design Condition		Normal Condition		Upset Condition		Testing Condition		Special Condition		Faulted Condition	
Component	Param. Per ASME Code III	Calc (psi)	Allow (psi)	Calc (psi)	Allow (psi)	Calc (psi)	Allow (psi)	Calc (psi)	Allow (psi)	Calc (psi)	Allow (psi)	Calc (psi)	Allow (psi)
Rod Travel Housing	P_m	6,288	16,100	N/A	N/A	N/A	N/A	7,815	16,110	N/A	N/A	7,621	38,640
	P_m+P_b	8,172	24,150	N/A	N/A	N/A	N/A	9,669	24,165	N/A	N/A	9,505	57,960
	P_m+P_b+Q	N/A	N/A	16,669	48,300	14,388	48,300	N/A	N/A	N/A	N/A	N/A	N/A
	$\sigma_1+\sigma_2+\sigma_3$	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	-14,654	64,400	N/A	N/A
Latch Housing	P_m	11,930	15,300	N/A	N/A	N/A	N/A	14,827	15,300	N/A	N/A	14,459	36,720
	P_m+P_b	15,659	22,950	N/A	N/A	N/A	N/A	18,556	22,950	N/A	N/A	18,188	55,080
	P_m+P_b+Q	N/A	N/A	17,431	45,900	16,395	45,900	N/A	N/A	N/A	N/A	N/A	N/A
	$\sigma_1+\sigma_2+\sigma_3$	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	15,056	61,200	N/A	N/A
Canopy	P_m	4,460	15,300	N/A	N/A	N/A	N/A	5,543	15,300	N/A	N/A	5,406	36,720
	P_m+P_b	6,844	22,950	N/A	N/A	N/A	N/A	7,927	22,950	N/A	N/A	7,790	55,080
	P_m+P_b+Q	N/A	N/A	45,504	45,900	38,164	45,900	N/A	N/A	N/A	N/A	N/A	N/A
	$\sigma_1+\sigma_2+\sigma_3$	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	5,439	61,200	N/A	N/A
Threaded Area	Pm (Shear)	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	3,314	9,180	N/A	N/A
	2x Shear	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	11,272	45,900	N/A	N/A
	P_m+P_b+Q	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	31,100	45,900	N/A	N/A
	Bell Mouthing Stress Intensity	N/A	N/A	14,136	17,000	11,069	17,000	N/A	N/A	N/A	N/A	N/A	N/A

Table 2.2.2.4-6 Unit 2 Lower Joint Stress Summary													
Lower Joint		Design Condition		Normal Condition		Upset Condition		Testing Condition		Special Condition		Faulted Condition	
Component	Param. Per ASME Code III	Calc (psi)	Allow (psi)	Calc (psi)	Allow (psi)	Calc (psi)	Allow (psi)	Calc (psi)	Allow (psi)	Calc (psi)	Allow (psi)	Calc (psi)	Allow (psi)
Latch Housing	P_m	12,380	15,300	N/A	N/A	N/A	N/A	15,386	21,375	N/A	N/A	15,005	36,720
	P_m+P_b	16,650	22,950	N/A	N/A	N/A	N/A	19,656	32,062	N/A	N/A	19,275	55080
	P_m+P_b+Q	N/A	N/A	16,921	45,900	15,228	45,900	N/A	N/A	N/A	N/A	N/A	N/A
	$\sigma_1+\sigma_2+\sigma_3$	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	15,560	61,200	N/A	N/A
Head Adaptor	P_m	7,343	16,100	N/A	N/A	N/A	N/A	9,126	16,100	N/A	N/A	8,900	38,640
	P_m+P_b	10,070	24,150	N/A	N/A	N/A	N/A	11,853	24,165	N/A	N/A	11,627	57,960
	P_m+P_b+Q	N/A	N/A	15,165	48,300	13,467	48,300	N/A	N/A	N/A	N/A	N/A	N/A
	$\sigma_1+\sigma_2+\sigma_3$	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	15,824	64,400	N/A	N/A
Canopy	P_m	9,345	15,300	N/A	N/A	N/A	N/A	11,614	15,300	N/A	N/A	11,326	36,720
	P_m+P_b	19,011	22,950	N/A	N/A	N/A	N/A	21,280	22,950	N/A	N/A	20,992	55,080
	P_m+P_b+Q	N/A	N/A	45,985(1)	45,900	37,154	45,900	N/A	N/A	N/A	N/A	N/A	N/A
	$\sigma_1+\sigma_2+\sigma_3$	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	28,702	61,200	N/A	N/A
Threaded Area	Pm (Shear)	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	4,103	9,180	N/A	N/A
	2x Shear	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	12,852	45,900	N/A	N/A
	P_m+P_b+Q	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	33,200	45,900	N/A	N/A
	Bell Mouthing Stress Intensity	N/A	N/A	13,733	17,000	9,720	17,000	N/A	N/A	N/A	N/A	N/A	N/A

Note:

1. This stress exceeds the allowable by .2%. This is considered insignificant due to the conservatism that the allowable is based on the design temperature of 650°F as opposed to the actual nodal temperature of 78°F. The ASME Code allowable stress intensity $3S_m$ is 60 ksi at 78°F and 45.9 ksi at 650°F.

2.2.2.5 Steam Generators and Supports

2.2.2.5.1 Regulatory Evaluation

The steam generators are described in Final Safety Analysis Report (FSAR) Sections 3.9N, 5.1 and 5.4.2. The steam generators supports are described in FSAR Section 5.4.14.2.2. Comanche Peak Nuclear Power Plant (CPNPP) Unit 1 uses four Westinghouse Model Δ76 steam generators. CPNPP Unit 2 uses four Westinghouse Model D-5 steam generators. The regulatory evaluation included in Licensing Report (LR) subsection 2.2.2 also applies to the steam generator and supports.

Current Licensing Basis

The current licensing basis provided in LR subsection 2.2.2 applies to the steam generator and its supports, with the following amplifications:

- FSAR Section 5.4.2.1.1 states, in part, that all pressure boundary materials used in the steam generator are selected and fabricated in accordance with the requirements of Section III of the American Society of Mechanical Engineers (ASME) Code. Table 5.2-1 of the FSAR provides ASME Boiler and Pressure Vessel (B&PV) Code Edition and Addenda applicable to the steam generators. A general discussion of materials specifications is given in FSAR Section 5.2.3, with types of materials listed in FSAR Tables 5.2-2 and 5.2-3.

- FSAR Section 5.4.2.1.1 describes the materials used for the steam generator tubes, divider plate, channel head and nozzles and support plates.

- FSAR Section 5.4.2.1.3 describes the steam generator tube support plates. The tube support plates are stainless steel and have quatrefoil or trefoil shaped holes; therefore, GL 88-02 does not impact CPNPP Unit 1 or 2.

- Steam generator design data is shown in FSAR Table 5.4-3. Code classifications for the steam generator components are provided in FSAR Section 3.2.

- FSAR Section 5.4.2.5 describes the possibility of degradation of tubes due to either mechanical or flow-induced excitation.

- The steam generators are inspected per the requirements of Section XI of the ASME B&PV Code and Technical Specifications. The steam generator quality assurance is described in FSAR Section 5.4.2.2 and summarized in FSAR Table 5.4-4.

- FSAR Section 5.4.14.2.2 describes the supports for each steam generator.

- FSAR Table 3.9N-1 summarizes RCS design transients, which apply to the steam generator, for normal, upset, emergency, faulted, and test conditions. FSAR Chapter 15 addresses component responses to various limiting design transients in more detail.

2.2.2.5.2 Technical Evaluation

2.2.2.5.2.1 Thermal-Hydraulic Evaluation (Unit 1)

Introduction

An analysis was performed to assure that the thermal-hydraulic performance CPNPP Unit 1 steam generators remain within acceptable bounds after the 3,628 MWt nuclear steam supply system (NSSS) SPU. The key thermal-hydraulic factors of interest include: (1) potential for tube dryout, (2) hydrodynamic stability, and (3) moisture carryover (MCO). The evaluation also includes an assessment of other secondary side operating characteristics such as steam generator mass, circulation ratio, and secondary side pressure drops. CPNPP Unit 1 has four Δ76 steam generators with eighteen 20-inch primary separators and a bank of six single-tier secondary separators (dryers).

Input Parameters, Assumptions, and Acceptance Criteria

Operating Conditions

The design parameters for the 3,628 MWt NSSS SPU are defined in LR Section 1.1. The design parameters support a vessel average temperature (T_{avg}) range of 574.2° to 589.2°F, feedwater temperature (T_{fw}) range of 390° to 450.3°F, and SGTP levels of 0 and 10 percent. The cases examined are labeled Case 1A/1B to Case 4A/4B and are summarized in Table 2.2.2.5-1. For Cases 1A to 4A, the feedwater temperature is 450.3°F; for Cases 1B to 4B, the feedwater temperature is 390°F.

The reference design parameters for this evaluation were established by considering a previously evaluated NSSS power output of 3,582 MWt in support of the Δ76 steam generators. The design parameters for the NSSS power output of 3,582 MWt support a vessel average temperature (T_{avg}) range of 574.2° to 589.2°F, feedwater temperature (T_{fw}) range of 390° to 448.6°F, and steam generator tube plugging (SGTP) levels of 0 and 10 percent. The cases examined are labeled Case 1A/1B to Case 4A/4B and are summarized in Table 2.2.2.5-2. For Cases 1A to 4A, the feedwater temperature is 448.6°F; for Cases 1B to 4B, the feedwater temperature is 390°F.

Acceptance Criteria

The relevant acceptance criteria for the 3,628 MWt NSSS SPU conditions are as follows:

- There is no local dryout on the tube wall.
- The damping factor for hydrodynamic instability evaluation is sufficiently negative.
- MCO remains acceptable below the design limit of 0.10 percent.

Description of Analyses and Evaluations

The GENF Code Version 1.1.5 was used to calculate the secondary side thermal-hydraulic characteristics at the 3,628 MWt NSSS SPU conditions. A three-dimensional flow field analysis for the secondary side of the steam generator was also performed using the ATHOS family of codes to examine the potential for local tube wall dryout. Local dryout on the tube wall is also evaluated using the correlation for the departure from nucleate boiling (DNB), which could result in excessive buildup of tube scale.

Method Discussion

Thermal-hydraulic conditions for the CPNPP Unit 1 steam generators are shown in Table 2.2.2.5-1.

The GENF code was used to calculate steady-state steam generator characteristics. The output from the GENF code includes various parameters such as primary temperatures, circulation ratio, steam flow rate, steam pressure, secondary side pressure drop, secondary fluid inventory, or damping factor. The GENF results are used to evaluate the acceptability of the steam generator performance with the SPU. The calculated operating conditions were utilized as input to the ATHOS family of codes for evaluating the margin-to-tube dryout for the worst-case operating conditions.

Moisture Carryover Evaluation

Excessive MCO may result in erosion-corrosion problems in the steam piping and/or steam turbine. Therefore, an MCO evaluation was performed for the 3,628 MWt NSSS SPU conditions.

MCO is essentially governed by three operating parameters: steam flow (power), steam pressure, and water level. The limiting operating case for MCO is Case 2A in Table 2.2.2.5-1, since this case results in the lowest outlet steam pressure that tends to produce maximum volumetric steam flow conditions.

Prediction of Secondary Side Mixture Quality at DNB

DNB is dependent on circulation ratio, steam pressure, and steam flow rate. Physically, if the liquid flow rate is not adequate, local dryout at the tube wall could take place. Since the liquid flow rate is dependent on circulation ratio, low circulation ratio means a low liquid flow rate through the tube bundle and a higher potential for local dryout. A lower steam pressure also increases the potential for DNB because it tends to produce a larger void fraction for the same steam quality. Therefore, a low reactor coolant temperature is more critical than a high coolant temperature, as the former results in a lower steam pressure. The combined effect of low values of circulation ratio, steam pressure, and coolant temperature tends to decrease the margin to DNB. It is, therefore, adequate to examine Case 2A because this case will have the highest potential for dryout, and, if this case is free of dryout, then all other cases are also free of dryout.

Results

The thermal-hydraulic evaluation of the CPNPP Unit 1 steam generators focused on the changes to the secondary side operating characteristics at the 3,628 MWt NSSS SPU conditions. The following is a summary of the results of the CPNPP Unit 1 steam generator at the analyzed 3,628 MWt SPU conditions:

- Damping factor

  The hydrodynamic stability of a steam generator is characterized by a damping factor. A negative value of this parameter indicates a stable unit. That is, small perturbations in thermal-hydraulic parameters (such as flow, pressure, or temperature) will die out rather than grow in amplitude. As indicated in Table 2.2.2.5-1, the damping factor remains at a high negative value, varying from [ ]^{a,c} hr^{-1} to [ ]^{a,c} hr^{-1}, for all uprated conditions. Consequently, the steam generators are expected to continue to operate in a hydrodynamically stable manner after the 3,628 MWt NSSS SPU.

- Tube dryout potential

  The DNB index increases with elevation in the tube bundle and peaks in the U-bend with the hot side exhibiting a higher index than the cold side. The highest value of the DNB index is equal to [ ]^{a,c} and occurs at a small locality in the U-bend. This indicates that the entire tube bundle is expected to be within the nucleate-boiling heat transfer regime and no local dryout is expected even for the worst post-uprate operating condition.

- Moisture carryover

  As part of the thermal-hydraulic evaluations to show acceptable operating characteristics at the 3,628 MWt NSSS SPU, an MCO evaluation was completed. Based on this analysis, MCO was estimated at [ ]^{a,c} percent by weight for nominal water level. With the water level at the upper end of the nominal operating range, MCO was estimated at [ ]^{a,c} percent. Although actual results may differ due to measurement uncertainty and normal variability, there is high confidence that MCO will remain below the design limit of 0.10 percent even for the worst post-uprate operating condition (that is, Case 2A with water level at the upper range).

- Circulation ratio

  The circulation ratio is a measure of the mixture flow in relation to the steam flow. It is primarily a function of the steam flow (power). Table 2.2.2.5-1 shows that circulation ratio may decrease by [ ]^{a,c} percent to [ ]^{a,c} percent at the 3,628 MWt NSSS SPU conditions when compared to similar (T_{avg}/T_{fw}/SGTP) cases shown in Table 2.2.2.5-2. The bundle mixture flow, given by circulation ratio times steam flow, may decrease by less than [ ]^{a,c} percent. The bundle flow is expected to be large enough to minimize accumulation of contaminants on the tubesheet and in the bundle. Therefore, no

significant effect on sludge accumulation or local concentrations is expected. As discussed previously, circulation ratio is sufficient to prevent local dryout on the tube wall.

- Steam generator secondary side mass

  Steam generator secondary side mass is affected by power, feedwater temperature, and primary side fluid temperature. As power increases, secondary side mass is decreased. For the same power level, a lower feedwater temperature results in a lower steam flow but increases circulation ratio and water mass. For the same power level, a higher primary side temperature results in a higher steam pressure, which reduces void fraction in the tube bundle and results in higher water mass. Moreover, for the same operating conditions, a higher tube plugging level results in a lower steam pressure, which increases void fraction in the tube bundle and, thus reduced water mass. For the various uprated conditions considered, the secondary side water mass is decreased by approximately [ ]^{a,c} percent relative to similar (T_{avg}/T_{fw}/SGTP) cases shown in Table 2.2.2.5-2. A small change of this magnitude will have no measurable effect on the processes related to liquid mass inventory in the bundle.

- Steam generator pressure drop

  The total steam generator pressure drop values shown in Table 2.2.2.5-1 represent the pressure drop from the feedwater nozzle inlet to the steam nozzle outlet. The total secondary side pressure drop after the 3,628 MWt NSSS SPU may increase by up to [ ]^{a,c} psi relative to similar (T_{avg}/T_{fw}/SGTP) cases shown in Table 2.2.2.5-2. This increase is small in relation to the total feedwater system pressure drop and should have no significant effect on the feedwater system operation.

- Average heat flux

  The average heat flux value will increase with power and tube plugging. With the 3,628 MWt NSSS SPU, increased total heat load is passed through the same bundle heat transfer area, increasing heat flux in proportion to the power increase. However, this increase in heat flux is acceptable since it will not lead to tube dryout as previously discussed.

Conclusion

All calculated thermal-hydraulic parameters are projected to remain within acceptable ranges for operation at the 3,628 MWt NSSS SPU conditions with tube plugging levels of up to 10 percent, as summarized in Table 2.2.2.5-1.

Unit 1 Steam Generator Structural Integrity Evaluation

Introduction

A structural integrity evaluation of the steam generator was performed for the CPNPP Unit 1 3,628 MWt NSSS SPU. Several operating and transient parameters were revised as a result of the 3,628 MWt NSSS SPU and this revised data is presented in the design specifications. CPNPP Unit 1 steam generator components were previously evaluated using operating and transient parameters that corresponded to a NSSS power output of 3,582 MWt. A structural evaluation of the CPNPP Unit 1 steam generator primary and secondary components was performed to determine the effect of the revised design transients associated with the 3,628 MWt NSSS SPU on the critical steam generator component stresses and fatigue usage.

Input Parameters, Assumptions, and Acceptance Criteria

The input parameters to the structural evaluation of the primary-and-secondary-side steam generator components associated with the 3,628 MWt SPU are included in LR Section 1.1. The stress and fatigue results from the original structural analyses of the steam generator components (that is, those for NSSS power output of 3,582 MWt) are also used as input assumptions.

A second input is the revised design transient data (LR subsection 2.2.6) that is used to determine appropriate scale factors, if necessary, to apply to the original results. The initial conditions for the design transients (that is, the steady-state operating parameters) are changed only slightly due to the 3,628 MWt NSSS SPU.

For this evaluation, the very small changes in the transient parameter (primary pressure, hot leg temperature, cold leg temperature, steam temperature and flow, and feedwater and auxiliary feedwater temperature and flow) do not impact the original stress and fatigue results calculated for an NSSS power output of 3,582 MWt.

The acceptance criteria for the structural evaluation of the CPNPP Unit 1 SPU are set forth in Section III of the ASME Boiler and Pressure Vessel Code (Reference 1) and remain the same as for the original structural analyses of the steam generator components.

Description of Analyses and Evaluations

Stress and fatigue results for the previously calculated 3,628 MWt NSSS SPU were calculated by applying scale factors, if necessary, to the previously calculated baseline stress and fatigue calculations and then compared to the stress and fatigue allowables of Reference 1.

Results

The stress and fatigue results in the primary-and-secondary-side steam generator components were evaluated to determine the impact of changes in some of the operating and design

transient parameters associated with the 3,628 MWt NSSS SPU. The primary and secondary components are discussed separately below.

Primary Side Components

The stress range and fatigue results for the following primary side steam generator components were evaluated for the 3,628 MWt NSSS SPU:

- Divider plates
- Tube-to-tubeplate welds
- Tubes
- Primary inlet and outlet nozzles
- Primary manways
- Primary chambers, tube plates, and lower shells

The minor differences in the plant design parameters associated with the NSSS SPU from 3,582 to 3,628 MWt will have negligible effect on the primary side design transients (see LR subsection 2.2.6). Therefore, the design transient data used in the structural evaluation of the primary side components remains applicable for the 3,628 MWt NSSS SPU. Consequently, the stress and fatigue results presented in the original analyses for these components remain valid for the 3,628 MWt NSSS SPU.

Secondary Side Components

The stress range and fatigue results for the following secondary side steam generator components were evaluated for the 3,628 MWt NSSS SPU:

- Secondary manways
- Steam outlet nozzles, elliptical heads, and upper shells
- Upper shell, transition cone, and lower shells
- 6-inch handholes and transition cone 4-inch inspection ports
- Lower shell 4-inch inspection ports
- 2.5-inch inspection ports
- Minor shell taps
- Upper internals components
- Lower internals components
- Auxiliary feedwater nozzles and thermal sleeves
- Main feedwater nozzles and thermal sleeves
- Feedrings, feedring supports, and spray nozzles

As a result of the 3,628 MWt NSSS SPU, the steam temperature curves for two upset transients (that is, loss of load and loss of power) were modified. All other secondary side steam temperature transients remain valid for the 3,628 MWt NSSS SPU.

For both the loss-of-load and the loss-of-power transients, the steam temperature variation between the uprate and baseline transient is less than [ ]^{a,c} °F. The thermal stress effects

associated with a temperature differential this small are considered negligible and are not included in determination of a scale factor. Further, the change in steam pressure associated with the steam temperature change was found to be small, such that a scale factor based on the maximum steam pressure during each transient is calculated to be 1.0. Because this scale factor is calculated to be 1.0, the stress range and fatigue results presented in original analyses remain valid for the 3,628 MWt NSSS SPU.

The design transients other than loss-of-load and loss-of-power transients remain applicable for the 3,628 MWt NSSS SPU. Consequently, the stress range and fatigue results presented in the original analyses for these components remain valid for the 3,628 MWt NSSS SPU.

For the 3,628 MWt NSSS SPU, the 100-percent power feedwater temperature is [ ]^{a,c} °F higher than for the baseline condition. The change to the feedwater temperature for the 3,628 MWt SPU also requires changes to the feedwater temperature transient curves.

Those steam generator secondary side components that are directly affected by modifications to the feedwater temperature transient data are the main feedwater nozzle and thermal sleeve, the feedring, feedring supports, and spray nozzles. Thermal stresses in these components arise from differential thermal expansion of dissimilar materials and from temperature gradients produced by feedwater fluid temperature on the inside of the component and by the secondary side fluid temperature on the outside of the components.

For those secondary side transients that did not change, the increase in feedwater temperature of [ ]^{a,c} °F will directly result in a reduction in the thermal gradient. For the loss-of-load and loss-of-power transients, it was shown previously that the secondary side temperature increases less than [ ]^{a,c} °F, such that an increase in feedwater temperature of [ ]^{a,c} °F will still result in a net reduction in the thermal gradient.

Since the thermal gradient produced by the feedwater/secondary fluid during the 3,628 MWt NSSS SPU is less than that for the 3,582 MWt NSSS power condition, the thermal stresses in the feedwater nozzle and thermal sleeve for the uprated condition are considered to be bounded by the current analyses. Consequently, the stress range and fatigue results presented in current analyses of these components remain valid for the 3,628 MWt NSSS SPU.

Conclusions

The results of the analyses and evaluations summarized herein demonstrate that the primary and secondary side steam generator components satisfy the structural requirements of Section III of the ASME Boiler and Pressure Vessel Code (Reference 1) for the CPNPP Unit 1 3,628 MWt NSSS SPU.

Unit 1 Steam Generator Design Pressure Differential Evaluation

Introduction

An analysis was performed to determine if the ASME B&PV Code 1989 Edition, no Addenda (Reference 1) limits on design primary to secondary pressure drop (ΔP) are exceeded for any of the applicable transient conditions for the CPNPP Unit 1 3,628 MWt NSSS SPU.

Input Parameters, Assumptions, and Acceptance Criteria

The design pressure limit for the primary to secondary side pressure differential is [ ]^{a,c} psi. The design pressure requirements for Class 1 equipment are defined in the applicable edition of the ASME Code Section III. For the CPNPP Unit 1 Δ76 steam generators, the 1989 Edition of the ASME Boiler and Pressure Vessel Code applies (Reference 1). The requirements are:

- Normal condition transients: Primary to secondary pressure gradient shall be less than the design limit of [ ]^{a,c} psi.

- Upset condition transients: If the pressure during an upset transient exceeds the design pressure limit, the stress limits corresponding to design conditions apply using an allowable stress intensity value of 110 percent of those defined for design conditions. In other words, in the case where all the upset condition pressure values are less than 110 percent of the design pressure values, no additional analysis is necessary. For the CPNPP Unit 1 steam generators, 110 percent of the design pressure limit corresponds to [ ]^{a,c} psi.

Description of Analyses and Evaluations

Two sets of transient parameters are defined. One corresponds to a high T_{avg} mode of operation and one corresponds to a low T_{avg} mode of operation for each of the NSSS operating limits analyzed, as summarized in Table 2.2.2.5-3. In addition, transient parameters are defined for two different tube plugging levels, 0 and 10 percent, for each of the operating limits. The pressure differentials across the primary to secondary side pressure boundary are calculated for the high T_{avg} and low T_{avg} full-power conditions corresponding to the 10-percent tube plugging level that bound the 0-percent plugging conditions.

Results

The analysis determined that for the high T_{avg} conditions, the maximum primary-to-secondary-side differential pressures are [ ]^{a,c} psi for normal operating condition transients and [ ]^{a,c} psi for upset condition transients. Both values are below the applicable CPNPP Unit 1 Δ76 steam generators' design pressure limits of [ ]^{a,c} psi for normal operating condition transients and [ ]^{a,c} psi for upset condition transients. These results are summarized in Table 2.2.2.5.-2.

For low T_{avg} conditions, the maximum primary-to-secondary-side differential pressures are [ ]^{a,c} psi for normal operating condition transients and [ ]^{a,c} psi for upset condition transients. These values are also below the applicable CPNPP Unit 1 Δ76 steam generators' design pressure limits of [ ]^{a,c} psi for normal operating condition transients and [ ]^{a,c} psi for upset condition transients. These results are also summarized in Table 2.2.2.5-4. The three power levels addressed are 3,475 MWt (current NSSS power), 3,582 MWt (Δ76 Steam Generator design NSSS power), and 3,628 MWt (uprated NSSS power).

Conclusion

The results of the analyses performed for the primary-to-secondary-side pressure differential for CPNPP Unit 1 are all below the applicable design pressure limits of [ ]^{a,c} psi and [ ]^{a,c} psi for normal and upset conditions, respectively.

Unit 1 Steam Generator Tube Integrity Evaluation

Introduction

The following evaluation presents an assessment of tube integrity at 3,628 MWt NSSS SPU conditions for the Unit 1 steam generators.

Input Parameters, Assumptions, and Acceptance Criteria

Input Parameters

Vessel outlet temperature (T_{hot}) CPNPP Unit 1 = 620.4°F for 0- to 10-percent steam generator tube plugging (SGTP).

Assumptions

It is assumed that the steam generator tube materials of construction will continue to perform similarly with plants of equal tube material specification.

Acceptance Criteria

The following acceptance criteria applies: The effect of operation at 3,628 MWt NSSS SPU conditions will not adversely affect tube integrity consistent with the performance criteria of NEI 97-06, Revision 2, "Steam Generator Program Guidelines" (Reference 2) as described in FSAR 5.4.2.2.

Description of Analyses and Evaluations

At the 1RF12 outage, the original D-4 steam generators were replaced with Δ76 steam generators. The Δ76 steam generator includes many design upgrades including:

- Thermally-treated Alloy 690 tubing
- Post-bending additional thermal treatment of Row 1 through 9 tubes

- Stainless steel tube support plates
- Trifoil tube hole design
- Hydraulically expanded tube-to-tubesheet joint design
- Reduced gap anti-vibration bars (AVBs)
- Two additional AVBs (6 total)
- High-efficiency moisture separators

These design improvements are expected to result in improved steam generator tube corrosion performance at temperatures associated with operation at 3,628 MWt NSSS SPU conditions. The evaluation performed for the uprate used a maximum predicted vessel outlet temperature of 620.4°F.

The Delta style steam generator has been used at other plants, such as Shearon Harris and V. C. Summer. The only difference between the steam generators at CPNPP Unit 1 and those identified above is an additional 1000 ft^2 of tubing heat transfer area.

The V. C. Summer Δ75 steam generators were installed in 1993. As of the most recent steam generator eddy current inspection (April 2005), eight tubes have been plugged at V. C. Summer. The Electric Power Research Institute (EPRI) Steam Generator Degradation Database (SGDD) lists three tubes plugged in September 1994 and five tubes plugged in October 2000. The plugging cause is listed as preventive for both outages in which tubes were plugged. The V. C. Summer steam generators have accumulated approximately 10.5 effective full-power years (EFPYs) since replacement. The EPRI SGDD lists the V. C. Summer and Shearon Harris hot leg operating temperatures at 619°F.

The Shearon Harris Δ75 steam generators were installed in the Fall of 2001. At the 2003 outage, three tubes were plugged due to foreign-object-induced tube wear at the top of tubesheet. The operating history of the Δ75 steam generators to date has shown little or no evidence of AVB wear, and no stress corrosion degradation. To date, there have been no industry reports of stress corrosion degradation of Alloy 690 thermally treated tubing material.

Tube integrity evaluations for the CPNPP Unit 1 Δ76 steam generators follow the guidance of the EPRI Steam Generator Integrity Assessment Guideline (Reference 3).

Conclusion

Based on this evaluation, the likelihood of stress corrosion degradation is not increased for operation of CPNPP Unit 1 at the 3,628 MWt NSSS SPU conditions.

Unit 1 Steam Generator Flow-Induced Vibration and Tube Wear Evaluation

Introduction

The flow-induced vibration (FIV) related impact of the CPNPP Unit 1 3,628 MWt NSSS SPU on the steam generator tubes was evaluated based on the current design basis analysis. It included the changes in the thermal-hydraulic characteristics of the secondary side of the steam generator resulting from the uprate. The effects of these changes on the fluid-elastic instability ratio, potential tube wear, and amplitudes of tube vibration due to turbulence have been addressed.

Input Parameters, Assumptions, and Acceptance Criteria

All inputs, assumptions, and acceptance criteria remained the same as the original vibration and wear analysis (which was performed for an NSSS power output of 3,582 MWt) except that the uprated condition results are used for thermal-hydraulic inputs.

Description of Analyses and Evaluations

Three FASTVIB runs were made to evaluate the effect on stability ratios from the uprated power conditions. The first run was for the U-bend, the second was for the hot leg, and the third was for the cold leg. FASTVIB uses a subset of the FLOVIB coding to automate multiple solutions for multiple geometries for the fluid-elastic and flow turbulence mechanisms. FLOVIB analyzes a single tube with a specific support condition. Additional FLOVIB analyses were performed for the postulated missing support cases to obtain limiting results for actual tube/AVB intersections, rather than for the nearest node spaced at 6-degree increments, and to obtain additional parameters needed to scale the semi-empirical wear calculations. The results from the current vibration and wear analysis were compared with the limiting uprated operating condition results and modified to account for changes in the secondary side operating conditions associated with the most limiting of the 3,628 MWt NSSS SPU conditions.

Results

The results are summarized below for the straight leg (see Table 2.2.2.5-6) and for the U-bend (see Table 2.2.2.5-7).

The straight leg and U-bend stability ratios were not impacted for normal operating conditions with all supports active. As a result, the analysis indicates that large amplitudes of vibration are not projected to occur due to the fluid-elastic mechanism while operating the steam generator at 3,628 MWt NSSS SPU conditions. As shown in Table 2.2.2.5-5, a [ ]^{a,c}-percent increase in stability ratio is seen when a single support is postulated to be inactive. It was determined in subsequent analyses that this increase did not produce any significant vibration or tube wear effects.

The largest root-mean-square (RMS) displacement due to turbulence for the 3,628 MWt NSSS SPU condition is still less than [ ]^{a,c} inches with a corresponding peak displacement of

[ ]^{a,c} inches. Corresponding stresses are small for both 100-percent power and uprated power operating conditions, that is, less than [ ]^{a,c} ksi RMS and less than [ ]^{a,c} ksi peak.

As in the original 3,582 MWt NSSS analysis, it was found that the vortex shedding mechanism was not a significant contributor to tube vibration since fluid-elastic excitation is the dominant excitation method for the tube bundle.

The potential for tube wear was addressed in the original 3,582 MWt NSSS analysis in both the straight leg and U-bend portions of the steam generator. These calculations were updated to reflect the operation of the steam generator at 3,628 MWt NSSS SPU conditions. The largest calculated wear depth for any single uncertainty using uprated operating conditions is [ ]^{a,c} inches. This represents about a [ ]^{a,c}-percent increase from the largest calculated wear depth for any single uncertainty from the original tube wear analysis. As was done in the original analysis, adding the combined square-root-of-the-sum-of-the-squares of the differences for each factor with the reference value results in a limiting value of [ ]^{a,c} inches. Thus, the bounding local wear depth value of [ ]^{a,c} inches from the original tube wear analysis can still be used for ASME Code evaluation of the tube. From these calculations, it can be concluded that although there may be an increase in the level of wear that would occur at the 3,628 MWt NSSS SPU conditions, the increased level is not significant.

In some models of steam generators, particular consideration is given to the potential for high cycle fatigue of U-bend tubes. This phenomenon has been observed in tubes with carbon steel support plates where denting or a fixed tube support condition has been observed in the uppermost plate. However, since the CPNPP Unit 1 steam generator tube support plates are manufactured from stainless steel, there is no potential for the necessary boundary conditions (that is, denting) to occur at the uppermost support plate. In the original analysis, it was determined that alternating stresses for all the FIV mechanisms are more than an order of magnitude below the lowest stress considered by the ASME design curves. They are also so low that the highest calculated off-nominal fatigue usage factor is less than [ ]^{a,c} even if the unexpected off-nominal environmental conditions are postulated to occur. The slightly increased uprated power stresses are still well below the lowest stress considered by the ASME design curves. Hence, the fatigue usage factor associated with FIV induced loadings while in the uprated operating condition is negligible, and fatigue degradation for FIV is not anticipated.

Conclusion

The analysis of the CPNPP Unit 1 steam generator demonstrates that the 3,628 MWt NSSS SPU conditions will not result in unacceptable FIV or tube wear. Limiting tube displacements, bending stresses, fatigue usage, and fluid-elastic stability ratios occur in the straight leg region, predominantly near the end of the tube lane, and are not changed much from the previous full-power conditions. Uprated power flow excitation in the U-bend region increases the fluid-elastic stability ratio by [ ]^{a,c} percent when all supports are included in the analyses, but this moderately higher potential is still less than the limiting overall value taken from the straight leg region. When multiple inactive support conditions are postulated for bounding tube wear calculations, the maximum calculated tube wear increases about [ ]^{a,c} percent, but the [ ]^{a,c}-inch local wear allowance is still sufficient for ASME Code evaluation of the tube.

Additional Steam Generator Hardware Evaluation

Mechanical repair hardware refers to components such as plugs and stabilizers that are installed in steam generator tubes to address tube degradation. These components were re-evaluated for the operating conditions and transients associated with the 3,628 MWt NSSS SPU. In anticipation of future tube plugging needs, both long and short [ ]^{a,c}-inch ribbed mechanical plugs were qualified for the 3,628 MWt NSSS SPU conditions. Additionally, weld plugs as originally installed in the shop are addressed. The other hardware item that has been evaluated for the 3,628 MWt NSSS SPU was the Westinghouse collared stabilizer. All repair components evaluated were found acceptable.

Unit 1 Steam Generator Chemistry Evaluation

Introduction

Water chemistry of both the primary and secondary sides in nuclear power plants is controlled to maximize the long-term availability of pressurized water reactor (PWR) plants. Primary water chemistry control can, and has been, effectively used to control radiation field buildup on ex-core surfaces also. Guidelines have been provided to utilities by EPRI for primary and secondary chemistry (References 4 and 5). In addition, other organizations such as NEI, have provided guidelines with respect to specific equipment (such as steam generators) that are incorporated into the EPRI guidelines. These documents form an industry consensus approach for chemistry programs that are embodied in and augmented by the plant-specific Strategic Water Chemistry Plans for the primary and secondary systems. Plant-specific strategies must consider plant design, ability to support chemistry control philosophies and targets, and cost-benefit tradeoffs. Uprates in power potentially affect water chemistry in the steam generator of the nuclear power plant because of changes in temperature and/or flow rates.

Input Parameters, Assumptions, and Acceptance Criteria

Input parameters include the operational parameters, EPRI primary and secondary chemistry guidelines (References 4 and 5), and strategic water chemistry plans for the primary and secondary sides at CPNPP.

A "Constant Elevated pH" chemistry program is followed for CPNPP Unit 1. This program was initiated on Unit 2 in Fuel Cycle 7 as a demonstration program and later implemented in CPNPP Unit 1 at Fuel Cycle 12. Morpholine and dimethylamine are used as the secondary amines. In addition, hydrazine is maintained at three times the condensate dissolved oxygen and at a minimum of 20 ppb to address upset conditions. The secondary chemistry program followed at CPNPP has resulted in steam generator degradation rates (for outside diameter stress corrosion cracking) at the lowest degradation class of preheat steam generators, startup iron values that are the lowest reported values in the industry, and the absence of measurable hideout return for Unit 1 reported in the Secondary Chemistry Program Evaluation for Unit 1 Cycle 11 Interim Summary. Chemistry performance for the Δ76 steam generators is expected to continue at this industry leading level.

There are no acceptance criteria for this evaluation.

Description of Analyses and Evaluations

EPRI guidelines recognize the difference in design and operating characteristics of nuclear plants and prescribes that each plant generate strategic water chemistry plans for the primary and secondary water chemistries. This allows for chemistry programs specifically tailored for each plant.

Results

CPNPP Unit 1 chemistry is controlled based upon strategies contained in their primary and secondary strategic chemistry programs.

Conclusion

No significant changes in the bulk chemistry of either the primary or secondary side are expected due to the 3,628 MWt NSSS SPU because the bulk chemistry will be continued to be controlled after the upgrade by plant procedures and specifications conforming to industry accepted guidelines and embodied in the CPNPP strategic water chemistry documents. In addition, the temperatures stated in uprated design parameters are in the range where other plants control bulk chemistry based on the same industry guidelines. Bulk water chemistry affects the chemistry in regions of the steam generators (or other components) where deleterious chemical species could accumulate and changes in temperatures and pressures can affect the superheat available and the amount of concentration possible in these regions. The potential for increased superheat in occluded environments underscores the importance of maintaining non-aggressive secondary side chemistry conditions.

References

1. ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Power Plant Components," 1989 Edition, no Addenda, The American Society of Mechanical Engineers, New York, New York.

2. NEI 97-06 Revision 2, "Steam Generator Program Guidelines," Nuclear Energy Institute, May 2005.

3. Technical Report 1012987, "Steam Generator Integrity Assessment Guidelines Revision 2," Electric Power Research Institute, July 2006.

4. EPRI PWR Primary Water Chemistry Guidelines, Volume 1, Revision 5, TR-105714-V1R5, March 2003.

5. EPRI Pressurized Water Reactor Secondary Water Chemistry Guidelines, Revision 6, 1008224, Final Report, December, 2004.

Table 2.2.2.5-1

CPNPP Unit 1 - Steam Generator Performance Characteristics with 3,628 MWt NSSS SPU

Case	Case 1A/1B Reduced Average Temperature, 0% Plugging	Case 2A/2B Reduced Average Temperature, 10% Plugging	Case 3A/3B Elevated Average Temperature, 0% Plugging	Case 4A/4B Elevated Average Temperature, 10% Plugging
Operating Conditions				
NSSS Power, MWt	3628	3628	3628	3628
Power, MWt/SG	907	907	907	907
Primary Temperature				
SG T_{hot}, °F	606.25	606.25	620.41	620.41
SG T_{cold}, °F	541.85	541.85	557.59	557.59
SG T_{avg}, °F[(1)]	574.05	574.05	589.00	589.00
Primary Flow, gpm	95700	95700	95700	95700
Feed Temperature, °F	450.3/390.0	450.3/390.0	450.3/390.0	450.3/390.0
Tube Fouling, hr-ft^2-°F/Btu	0.00011	0.00011	0.00011	0.00011
Tube Plugging, %	0	10	0	10
Operating Characteristics				

a,c

Table 2.2.2.5-2

CPNPP Unit 1 - Steam Generator Performance Characteristics with 3,582 MWt NSSS SPU

Case	Case 1A/1B Reduced Average Temperature, 0% Plugging	Case 2A/2B Reduced Average Temperature, 10% Plugging	Case 3A/3B Elevated Average Temperature, 0% Plugging	Case 4A/4B Elevated Average Temperature, 10% Plugging
Operating Conditions				
NSSS Power, MWt	3582	3582	3582	3582
Power, MWt/SG	895.5	895.5	895.5	895.5
Primary Temperature				
SG T_{hot}, °F	605.87	605.87	620.09	620.09
SG T_{cold}, °F	542.23	542.23	558.01	558.01
SG T_{avg}, °F[1]	574.05	574.05	589.05	589.05
Primary Flow, gpm	95700	95700	95700	95700
Feed Temperature, °F	448.6/390.0	448.6/390.0	448.6/390.0	448.6/390.0
Tube Fouling, hr-ft^2-°F/Btu	0.00011	0.00011	0.00011	0.00011
Tube Plugging, %	0	10	0	10
Operating Characteristics				

a,c

Table 2.2.2.5-3 Bounding Parameters						
	CPNPP Unit 1 3,475 MWt Power		**CPNPP Unit 1 3,582 MWt Power**		**CPNPP Unit 1 3,628 MWt Power**	
PCWG Parameters	**Low T_{avg}**	**High T_{avg}**	**Low T_{avg}**	**High T_{avg}**	**Low T_{avg}**	**High T_{avg}**
NSSS Power, MWt	3,475	3,475	3,582	3,582	3,628	3,628
RCS Pressure, psia	2,250	2,250	2,250	2,250	2,250	2,250
Vessel Average Temperature, °F	574.2	589.2	574.2	589.2	574.2	589.2
Steam Temperature, °F	529.0	545.1	527.5	543.8	526.9	543.1
Steam Pressure, psia	877[(1)]	1,004[(1)]	867[(2)]	993[(2)]	862[(3)]	988[(3)]
Tube Plugging, %	10	10	10	10	10	10

Notes:

1. 14 psi steam generator internal pressure drop incorporated.
2. 15 psi steam generator internal pressure drop incorporated.
3. 16 psi steam generator internal pressure drop incorporated.

Table 2.2.2.5-4

Summary of ΔP for Comanche Peak Unit 1 Steam Generator Program

Case	Limiting Transient	Transient Condition	ΔP (psi) a,c	Allowable (psi) a,c
High T_{avg}, 10% Tube Plugging (3,475 MWt NSSS Power)	Feedwater Cycling	Normal		
	Excessive Feedwater	Upset		
High T_{avg}, 10% Tube Plugging (3,582 MWt NSSS Power)	Feedwater Cycling	Normal		
	Excessive Feedwater	Upset		
High T_{avg}, 10% Tube Plugging (3,628 MWt NSSS Power)	Feedwater Cycling	Normal		
	Excessive Feedwater	Upset		
Low T_{avg}, 10% Tube Plugging (3,475 MWt NSSS Power)	Unit Loading (15 - 100%)	Normal		
	Loss of Load	Upset		
Low T_{avg}, 10% Tube Plugging (3,582 MWt NSSS Power)	Unit Loading (15 - 100%)	Normal		
	Loss of Load	Upset		
Low T_{avg}, 10% Tube Plugging (3,628 MWt NSSS Power)	Unit Loading (15 - 100%)	Normal		
	Loss of Load	Upset		

Table 2.2.2.5-5

Comparison of Fluid-Elastic Stability Ratios from Current Power and Uprated Power Conditions for Limiting U-bend Tubes for Active and Postulated Inactive Single Support Locations

a,c

Table 2.2.2.5-6

Summary Vibration Analysis Results for Straight Leg Region

a,c

Peripheral Tubes									

Table 2.2.2.5-7

Summary Vibration Analysis Results for U-Bend Region

a,c

2.2.2.5.2.2 Thermal-Hydraulic Evaluation (Unit 2)

Introduction

A thermal-hydraulic analysis was performed to assure that the CPNPP Unit 2 steam generators performance remains within acceptable bounds after the 3,628 MWt nuclear steam supply system (NSSS) SPU. The key thermal-hydraulic factors of interest include: (1) potential for tube dryout, (2) hydrodynamic stability, and (3) MCO. The evaluation also includes an assessment of other secondary-side operating characteristics such as steam generator mass, circulation ratio, and secondary side pressure drops. CPNPP Unit 2 has four Model D-5 steam generators with twenty 20-inch diameter moisture separators.

Input Parameters, Assumptions, and Acceptance Criteria

<u>Operating Conditions</u>

The Model D-5 steam generators have been evaluated for operation at the uprated power conditions. The steam generator evaluations include an NSSS power of 3,628 MWt, SGTP in the range of 0 to 10 percent, a vessel average temperature (T_{avg}) range of 574.2° to 589.2°F, and a feedwater temperature (T_{fw}) range of 390° to 450.3°F.

The reference case for this evaluation was a previously implemented 3,475 MWt NSSS SPU, shown on Table 2.2.2.5-8.

The design parameters for the 3,628 MWt NSSS SPU are defined in LR Section 1.1. Cases 1 through 4 are evaluated for the 3,628 MWt NSSS SPU and correspond to the eight operating cases shown in LR Section 1.1. The GENF input data for all eight cases are summarized in Table 2.2.2.5-9. The reactor coolant vessel average temperature (T_{avg}) shown in the uprate design parameters are used as input to the GENF code for thermal-hydraulic evaluations.

<u>Acceptance Criteria</u>

The relevant acceptance criteria for the 3,628 MWt NSSS SPU conditions are as follows:

- There is no local dryout on the tube wall.
- The damping factor for hydrodynamic instability evaluation is sufficiently negative.
- MCO remains acceptable below the design limit of 0.25 percent. A higher MCO limit of [ ]^{a,c} percent is acceptable for a short duration. This limit is based on the upper limit for erosion-corrosion.

Description of Analyses and Evaluations

The GENF code, Version 1.0.0, was used to calculate the secondary side thermal-hydraulic characteristics for the 3,628 MWt NSSS SPU conditions. A three-dimensional-flow field analysis

for the secondary side of the steam generator was also performed using the ATHOS family of codes to examine the potential for local tube wall dryout. Local dryout on the tube wall is also evaluated using the correlation for the DNB, which could result in excessive buildup of tube scale.

Method Discussion

Thermal-hydraulic conditions for the CPNPP Unit 2 steam generators are shown in Table 2.2.2.5-9.

The GENF code was used to calculate steady-state steam generator characteristics. The output from the GENF code includes various parameters such as primary temperatures, circulation ratio, steam flow rate, steam pressure, secondary side pressure drop, secondary fluid inventory, or damping factor. The GENF results are used to evaluate the acceptability of the steam generator performance with the SPU. The calculated operating conditions were utilized as input to the ATHOS family of codes for evaluating the margin-to-tube dryout for the worst-case operating conditions.

Moisture Carryover Evaluation

Excessive MCO may result in erosion-corrosion problems in the steam piping and/or steam turbine. Therefore, an MCO assessment was performed for the 3,628 MWt NSSS SPU conditions.

MCO is essentially governed by three operating parameters: steam flow (power), steam pressure, and water level. The limiting operating case for MCO is Case 2A in Table 2.2.2.5-9, since this case results in the lowest outlet steam pressure which tends to produce maximum volumetric steam flow conditions.

Prediction of Secondary Side Mixture Quality at DNB

DNB is dependent on circulation ratio, steam pressure, and steam flow rate. Physically, if the liquid flow rate is not adequate, local dryout at the tube wall could take place. Since the liquid flow rate is dependent on circulation ratio, low circulation ratio means a low liquid flow rate through the tube bundle and a higher potential for local dryout. A lower steam pressure also increases the potential for DNB because it tends to produce a larger void fraction for the same steam quality. Therefore, a low reactor coolant temperature is more critical than a high coolant temperature, as the former results in a lower steam pressure. The combined effect of low values of circulation ratio, steam pressure, and coolant temperature tends to decrease the margin to DNB. It is, therefore, adequate to examine Case 2A because this case will have the highest potential for dryout, and, if this case is free of dryout, then all other cases are also free of dryout.

Results

The thermal-hydraulic evaluation of the CPNPP Unit 2 steam generators focused on the changes to the secondary side operating characteristics at the 3,628 MWt NSSS SPU conditions. The following is a summary of the results of the CPNPP Unit 2 steam generator at the analyzed 3,628 MWt SPU conditions:

- Damping factor

  The hydrodynamic stability of a steam generator is characterized by a damping factor. A negative value of this parameter indicates a stable unit. That is, small perturbations in thermal-hydraulic parameters (such as flow, pressure, or temperature) will die out rather than grow in amplitude. As indicated in Table 2.2.2.5-9, the damping factor remains at a high negative value, varying from [ ]^{a,c} hr^{-1} to [ ]^{a,c} hr^{-1}, for all uprated conditions. Consequently, the steam generators are expected to continue to operate in a hydrodynamically stable manner after the 3,628 MWt NSSS SPU.

- Tube dryout potential

  The DNB index increases with elevation in the tube bundle and peaks in the U-bend with the hot side exhibiting a higher index than the cold side. The highest value of the DNB index is equal to [ ]^{a,c} and occurs at a small locality in the U-bend. This indicates that the entire tube bundle is expected to be within the nucleate boiling heat transfer regime and no local dryout is expected even for the worst post-uprate operating condition.

- Moisture carryover

  As part of the thermal-hydraulic evaluations to show acceptable operating characteristics at the 3,628 MWt NSSS SPU, an MCO assessment was completed. All of the MCO values except for Case 1A and Case 2A (Table 2.2.2.5-9) remain below the design limit of 0.25 percent. However, a higher MCO limit of 0.5 percent is acceptable for a short duration. This limit is based on the upper limit for erosion-corrosion. While the predicted value is above the limit, it still remains below the level of concern for the downstream equipment. Experience shows that measured MCO tends to be less than predicted.

- Circulation ratio

  The circulation ratio is a measure of the mixture flow in relation to the steam flow. It is primarily a function of the steam flow (power). Table 2.2.2.5-9 shows that circulation ratio varies from [ ]^{a,c} % to [ ]^{a,c} % at the 3,628 MWt NSSS SPU conditions relative to the reference case. The bundle mixture flow, given by circulation ratio times steam flow, varies from [ ]^{a,c} % to [ ]^{a,c} %. The bundle flow is expected to be large enough to minimize accumulation of contaminants on the tubesheet and in the bundle. Therefore, no significant effect on sludge accumulation or local concentrations is

expected. As discussed previously, circulation ratio is sufficient to prevent local dryout on the tube wall.

- Steam generator secondary side mass

  Steam generator secondary side mass is affected by power, feedwater temperature, and primary side fluid temperature. As power increases, secondary side mass is decreased. For the same power level, a lower feedwater temperature results in a lower steam flow but increases circulation ratio and water mass. For the same power level, a higher primary side temperature results in a higher steam pressure, which reduces void fraction in the tube bundle and results in higher water mass. Moreover, for the same operating conditions, a higher tube plugging level results in a lower steam pressure, which increases void fraction in the tube bundle and thus reduces water mass. For the various uprated conditions considered, the secondary-side water mass may vary from [ ]^{a,c} percent to [ ]^{a,c} percent, relative to the current reference case. A change of this magnitude will have no measurable effect on the processes related to the void in the bundle.

- Steam generator pressure drop

  The total steam generator pressure drop values shown in Table 2.2.2.5-9 represent a pressure drop from the feedwater nozzle inlet to the steam nozzle outlet. The inclusion of the steam generator blowdown slightly increases feedwater flow, which in turn increases the pressure drop in the inlet feedwater chamber and downcomer. The total secondary side pressure drop (from the feedwater nozzle inlet to the steam nozzle outlet) after the 3,628 MWt SPU may increase by up to [ ]^{a,c} psi. This increase is small relative to the total feedwater system pressure drop and will have no significant effect on the feedwater system operation.

- Average heat flux

  The average heat flux value will increase with power and tube plugging. With the 3,628 MWt NSSS SPU, increased total heat load is passed through the same bundle heat transfer area, increasing heat flux in proportion to the power increase. However, this increase in heat flux is acceptable since it will not lead to tube dryout as previously discussed.

Conclusion

All calculated thermal-hydraulic parameters are projected to remain within acceptable ranges for operation at the 3,628 MWt NSSS SPU conditions with tube plugging levels of up to 10 percent, as summarized in Table 2.2.2.5-9.

Unit 2 Steam Generator Structural Integrity Evaluation

Introduction

The structural integrity evaluation of the CPNPP Unit 2 Model D-5 steam generators was performed for a 3,628 MWt NSSS SPU. The stresses, stress intensity ranges, and fatigue usage factors in the steam generators for the SPU range of conditions were determined by reconciling the original design basis analyses against the SPU conditions provided in the uprate design parameters in LR Section 1.1 and NSSS design transients in LR subsection 2.2.6.

The scope of the reconciliation included the entire steam generator pressure boundary, internal and external pressure boundary attachments, and all internal components. Formal reconciliations were performed for the divider plate, tubesheet and shell junction, tube-to-tubesheet weld, tubes, feedwater nozzle, auxiliary feedwater nozzle, secondary manway bolts, steam nozzle, secondary-side wrapper support system components, and blowdown pipe. These components were selected for formal reconciliation because they have the limiting stress levels in the Unit 2 Model D-5 steam generators.

Input Parameters, Assumptions, and Acceptance Criteria

Input Parameters

The input to the evaluations for the SPU involve design primary to secondary pressure drops (ΔPs) that have been calculated for the key transients.

Stress results were extracted from the original design analysis. The applicable stress reports address the original design basis analysis. This analysis was later revised to address changes to the transient cycles in a revision of the generic design specification as well as the effects of running the feedwater/auxiliary feedwater systems under split-flow conditions. An evaluation of the latter shows the net effect on the fatigue usage to be negligible for most steam generator components. The change to the feedwater systems was also negligible due to changes to the line operations and changes to the transients evaluated for the auxiliary feedwater system. The application of the changes in stress due to the SPU is, therefore, conservatively applied to the original analysis results.

Assumptions

- The enveloping transient for the particular item being evaluated is identified in the various referenced stress reports remains the enveloping transient. Analysis to address revisions to the general design specification, and to address the feedwater/auxiliary feedwater flow split operation, stratification, and striping has been reviewed. The fatigue usage reported in the original stress reports generally remains applicable since there was little (a maximum [ ]^{a,c} change in fatigue usage) or no change in the fatigue results. Only the feedwater and auxiliary feedwater nozzles were impacted by the changes. These components were reanalyzed and the resulting fatigue usage decreased at some locations while increasing at other nozzle locations. The maximum

fatigue usage factor; therefore, occurs in the original analysis. This is attributed to refinements in the analysis model and removing conservatism from the analysis. Therefore, it is appropriate to perform this evaluation on the original analysis using the limiting fatigue usage as the basis to evaluate the effect of the uprate.

- The thermal stress differentials that would occur due to the SPU are considered to be of a negligible magnitude for each of the primary side components. Therefore, for the components evaluated, the scale factors calculated based on pressure are applied conservatively to those stresses that are a combination of both thermal and pressure stresses in the calculation of stress ranges and fatigue usage.

- Temperature effects for the secondary side components are accounted for through the consideration of the change in pressure and the application of the scaling factors applied as a result of the pressure evaluation, with the exception of the feedwater and auxiliary feedwater nozzles. The change in the minimum feedwater temperature, from 440° to 390°F, is accounted for and the change in the stress range due to the change is considered in the fatigue analysis.

- The steam pressure for the reference case is based on the P_{stm} value of [ ]^{a,c} psia. The maximum ΔP for the low T_{avg} uprate case for normal transients is equal to [ ]^{a,c} psi and is less than the maximum allowable value of [ ]^{a,c} psi. The maximum ΔP for the upset transient is equal to [ ]^{a,c} psi and is less than the maximum allowable value of [ ]^{a,c} psi.

- The stresses of the primary side components are proportional to the ΔP between the primary and secondary side. The scale factor, equal to $\Delta P_{uprate}/\Delta P_{reference}$, is applied to the reference analysis stresses when ΔP_{uprate} is greater than $\Delta P_{reference}$.

Acceptance Criteria

The acceptance criteria for the structural analysis for SPU conditions are the continued compliance with the current steam generator design basis analysis. For the structural evaluation of the pressure boundary components, the acceptance criteria from the ASME Code, Section III, Subsection NB for Class 1 components continued to remain applicable (Reference 1). Excessive plastic deformation is prevented by limits on the acceptable primary stresses. Plastic instability and incremental collapse are prevented by limits on the acceptable primary-plus-secondary stresses. High-strain, low-cycle fatigue is prevented by limits on the total stresses and their cycles. Satisfaction of these limits demonstrates continued compliance with the current design acceptance criteria and, therefore, the adequacy of the steam generator design for operation at the uprate conditions for the remainder of the 40-year design life.

The steam generator internal components, other than the U-tubes, are not part of the pressure boundary and, therefore, are not governed by the ASME Code. However, ASME Code Section III, Subsections NB and NF were adopted as guidelines for performing the structural analysis of these components (Reference 1). These components were reviewed and it was

determined that they satisfy the ASME Code requirements for components not requiring an analysis for cyclic operation. As a result, a fatigue analysis was not performed for the internals.

In cases in which the fatigue usage factor exceeds 1.0 for the uprate condition, the fatigue usage was evaluated based on 15 years of life prior to the uprate and the remaining 25 years at SPU conditions (based on the uprate occurring in 2008).

Description of Analyses and Evaluations

The primary stresses for design, test, emergency, and faulted conditions are not affected due to the uprating. The stress and fatigue are affected for normal/upset conditions due to the uprate and are evaluated in this calculation.

The stresses for the components exposed to both the primary and the secondary side of the steam generator are proportional to the differential pressure between the primary and secondary side. The current stresses for the various transient loading conditions have been scaled up by the ratio of the primary-to-secondary side differential pressure for the baseline and SPU conditions.

The stress range and the fatigue evaluations are calculated by applying the scaling factors to the current stress and fatigue calculations.

The secondary side components are exposed to the secondary side pressure. For secondary side components, it is assumed that the stress range increases due to the additional stress caused as the result of reduced steam pressure at 100-percent power. The additional stress is added to the stress range to evaluate the revised stress range and fatigue for the secondary side components. These additional pressure stresses are added to the alternating stresses in the fatigue evaluation and the revised fatigue usage is calculated. In the case of the feedwater and auxiliary feedwater nozzles, the effect of a lower maximum feedwater temperature is also taken into account and the fatigue usage is adjusted to consider this lower temperature.

Results

The stress and fatigue results in the primary-and-secondary-side steam generator components were evaluated to determine the impact of changes in some of the operating and design transient parameters associated with the 3,628 MWt NSSS SPU. The primary and secondary side components are discussed separately below.

Primary side components evaluated:

- Divider plates
- Tube-to-tubesheet welds
- Tubesheet and shell junctions
- Tubes

Secondary side components evaluated (the following are the components which exhibit the highest stress levels; other components are bounded by stress levels in these components):

- Feedwater nozzles
- Auxiliary feedwater nozzles
- Secondary side manway bolts
- 6-inch handholes and transition cone 4-inch inspection ports
- Steam nozzles
- Wrapper support system components
- Blowdown pipes

The normal/upset maximum stress intensity range and fatigue usage factors are provided in Tables 2.2.2.5-10 and -11 for primary side and secondary side components, respectively.

- The primary-plus-secondary stress range of the divider plate and tube-to-tubesheet weld exceeded the $3S_m$ limit and the plastic analyses that were performed in the original analyses. For the other primary and secondary components that exceeded the $3S_m$ limit, a simplified elastic-plastic analysis was performed in the original analysis as per the ASME Code.

- The fatigue usage of all primary side components remains ≤ 1.0.

- The fatigue usage for the secondary side components, with the exception of the secondary manway bolts, remains less than 1.0 and is acceptable. In the case of the secondary manway bolts, the baseline fatigue required the replacement of the bolts after 20 years of service. As the result of the uprate, the secondary manway bolts must be replaced on an 18-year schedule. This measure reduces the fatigue usage resulting from the uprate to below 1.0 and meets ASME Code limits.

The stress intensities of the wrapper support components are very small. As a result, the maximum fatigue usage for some components is less than 0.1, while for others, the fatigue usage is insignificant, with the stress ranges calculated below the endurance limit. The effect of the uprate on these components is insignificant.

Conclusions

The results of the analyses and evaluations summarized herein demonstrate that the primary and secondary side steam generator components satisfy the structural requirements of Section III of the ASME Boiler and Pressure Vessel Code (ASME B&PV) (Reference 1) for the CPNPP Unit 2 3,628 MWt NSSS SPU.

Unit 2 Steam Generator Design Pressure Differential Evaluation

Introduction

An analysis was performed to determine if the ASME B&PV Code, 1971 plus addenda through Summer 1973 (Reference 1) limits on design primary-to-secondary pressure drop (ΔP) are exceeded for any of the applicable transient conditions for the CPNPP Unit 2 3,628 MWt NSSS SPU.

Input Parameters, Assumptions, and Acceptance Criteria

The design pressure limit for the primary to secondary side pressure differential is [ ]^{a,c} psi. The design pressure requirements for Class 1 equipment are defined in the applicable edition of the ASME Code Section III. For the CPNPP Unit 2 steam generators, the 1971 Edition of the ASME Boiler and Pressure Vessel Code applies (Reference 1). The requirements are:

- Normal condition transients: Primary-to-secondary pressure gradient shall be less than the design limit of [ ]^{a,c} psi.
- Upset condition transients: If the pressure during an upset transient exceeds the design pressure limit, the stress limits corresponding to design conditions apply using an allowable stress intensity value of 110 percent of those defined for design conditions. In other words, in the case where all the upset condition pressure values are less than 110 percent of the design pressure values, no additional analysis is necessary. For the CPNPP Unit 2 steam generators, 110 percent of the design pressure limit corresponds to [ ]^{a,c} psi.

Description of Analyses and Evaluations

Two sets of transient parameters are defined. One corresponds to a high T_{avg} mode of operation and one corresponds to a low T_{avg} mode of operation for each of the NSSS operating limits analyzed, as summarized in Table 2.2.2.5-12. In addition, transient parameters are defined for two different tube plugging levels, 0 and 10 percent, for each of the operating limits. The pressure differentials across the primary-to-secondary-side pressure boundary are calculated for the high T_{avg} and low T_{avg} full-power conditions corresponding to the 10-percent tube plugging level which bound the 0-percent plugging conditions.

Results

The analysis determined that the maximum primary-to-secondary-side differential pressures during normal operating condition transients would be [ ]^{a,c} psi for the high T_{avg} temperature condition and [ ]^{a,c} psi for the low T_{avg} temperature condition. For upset condition transients, the maximum primary-to-secondary-side differential pressure would be [ ]^{a,c} psi for the high T_{avg} condition and [ ]^{a,c} psi for the low T_{avg} temperature condition. These results are summarized in Table 2.2.2.5-13.

Conclusion

The results of the analyses performed for the primary-to-secondary-side pressure differential for CPNPP Unit 2 are all below the applicable design pressure limits of [ ]^{a,c} psi and [ ]^{a,c} psi for normal and upset conditions, respectively.

Unit 2 Steam Generator Tube Integrity Evaluation

Introduction

The following evaluation presents an assessment of tube integrity at 3,628 MWt NSSS SPU conditions for the CPNPP Unit 2 steam generators.

Input Parameters, Assumptions, and Acceptance Criteria

Input Parameters

Vessel outlet temperature (T_{hot}) CPNPP Unit 2 = 620.4°F for 0- to 10-percent SGTP.

Assumptions

It is assumed that the steam generator tube materials of construction will continue to perform similarly with plants of equal tube material specification.

Acceptance Criteria

The following acceptance criteria applies: The effect of operation at 3,628 MWt NSSS SPU conditions will not adversely affect tube integrity consistent with the performance criteria specified by NEI 97-06, Revision 2, "Steam Generator Program Guidelines" (Reference 2).

Description of Analyses and Evaluations

The CPNPP Unit 2 steam generators are Model D-5. This steam generator design uses thermally-treated Alloy 600 tubing, hydraulic tube expansion within the tubesheet, stainless steel quatrefoil style tube hole tube support plates (TSPs), and a stainless steel flow distribution baffle. The Row 1 through Row 10 U-bends received an additional thermal treatment after bending. Considering the tube material properties for the post-bending thermal treatment, the tube residual stresses due to bending should be reduced to near straight leg values. Therefore, the potential for primary water stress corrosion cracking (PWSCC) in the small radius U-bends is significantly reduced. The Model D-5 steam generator is a preheater design, and as such, one mechanism experienced by this steam generator design is tube wear at preheater baffles. To date, no stress corrosion degradation has been reported at CPNPP Unit 2, and approximately 268 tubes are affected by AVB wear. At the last eddy current inspection (2RF08) the deepest reported AVB wear depth was 37 percent through wall (TW), which is less than the Technical Specification repair limit of 40-percent TW. At the 2RF08 outage, all tubes containing AVB wear indication depths of greater than or equal to 35-percent TW were administratively

plugged, as no eddy current inspection was planned for the 2RF09 outage. As of the most recent outage (2RF09), CPNPP Unit 2 has accumulated 11.6 effective full-power years (EFPYs).

For the 3,628 MWt NSSS SPU conditions, the CPNPP Unit 2 hot leg design temperature is 620.4°F. Therefore, the impact of uprating upon stress corrosion performance is considered negligible. For all previous outages, and for future outages, tube integrity for the CPNPP Unit 2 Model D-5 steam generators will follow the guidance of the EPRI Steam Generator Integrity Assessment Guideline.

The only degradation mechanism whose impact upon steam generator tube integrity can be quantified prior to implementation of the uprating is tube wear due to interaction with structures. The Model D-5 steam generators at CPNPP Unit 2 currently experience minimal tube wear at AVB intersections; no baffle plate wear was reported at the 2RF08 outage. The impact of uprating upon steam generator tube wear mechanisms has been evaluated. This evaluation has established that the product of fluid density times the square of velocity (ρV^2) could increase by [ ]^{a,c} percent for the uprated condition. AVB wear rates are expected to increase, but by a value much less than the ρV^2 change. This is due to the fact that wear mechanisms involve a constant energy dissipation, or work rate, and as the depth of the wear indication is increased (due to tube geometry), the volume of removed material is greatly increased. Therefore, as the removed volume to achieve a specified depth is increased exponentially and work rate remains constant, the wear scar depth growth rate as a function of time actually decreases with increasing indication depth. This has been observed in the CPNPP Unit 2 AVB wear growth data in that long term trending has indicated reduced growth rates with increasing accumulated EFPYs.

Conclusion

Based on this evaluation, the following conclusions are supported for operation of CPNPP Units at 3,628 MWt NSSS SPU conditions:

- The likelihood of stress corrosion degradation is not increased.

- Growth rates of currently observed tube wear mechanisms at CPNPP Unit 2 may be slightly increased. However, the magnitude of this increase is sufficiently small so that steam generator tube integrity performance criteria defined by Reference 1 will not be challenged.

Steam Generator Flow-Induced Vibration and Tube Wear Evaluation

Introduction

The FIV related impact of the CPNPP Unit 2 3,628 MWt NSSS SPU on the steam generator tubes was evaluated based on the current design basis analysis. It included the changes in the thermal-hydraulic characteristics of the secondary side of the steam generator resulting from the

uprate. The effects of these changes on the fluid-elastic instability ratio, potential tube wear, and amplitudes of tube vibration due to turbulence have been addressed.

Input Parameters, Assumptions, and Acceptance Criteria

All inputs, assumptions, and acceptance criteria remained the same as the current vibration and wear analysis, except that the SPU ATHOS/VGVB results were used for thermal-hydraulic inputs. The original vibration analysis demonstrated that the maximum fluid-elastic stability ratio for the expected tube support conditions was less than the allowable limit of 1.0. The original tube vibration analysis also determined that negligible tube responses occurred due to the vortex shedding mechanism. The amplitudes of vibration due to turbulence were also found to be reasonably small, with maximum displacements found to be on the order of [ ]^{a,c} mil. The maximum expected tube wear that could occur over 40 years of operation was calculated to be less than [ ]^{a,c} mils.

Description of Analyses and Evaluations

The results from the original vibration and wear analysis were modified to account for changes in the secondary-side operating conditions associated with the most limiting of the SPU conditions. The previously established values of fluid-elastic instability, turbulent amplitudes of vibration, and tube wear were modified for the SPU conditions.

Preheater tube wear was modified to address the effect of increased flow through the main feedwater nozzle into the preheater and the resultant impact on tube wear.

Results

The straight leg stability ratios were not impacted. The stability ratios for U-bend conditions increased from [ ]^{a,c} to [ ]^{a,c} which is still less than the allowable limit of 1.0. As a result, the analysis indicates that large amplitudes of vibration are not projected to occur due to the fluid-elastic mechanism while operating the steam generators at SPU conditions.

The maximum displacement values for turbulence excitation calculated in the original analysis were modified to account for SPU conditions. For the most limiting tube support condition, it was determined that the turbulence-induced displacement could increase from [ ]^{a,c} mil to [ ]^{a,c} mils. Displacements of this magnitude are not sufficient to produce tube-to-tube contact.

As in the original analysis, the vortex shedding mechanism was not a significant contributor to tube vibration at SPU conditions since fluid-elastic excitation is the dominant excitation mechanism for the tube bundle.

The potential for tube wear was addressed in the original analysis in both the straight-leg and U-bend portions of the steam generators. These calculations were updated to reflect the operation of the steam generators at SPU conditions. These calculations determined that the level of tube wear that could occur would increase by [ ]^{a,c} percent, from [ ]^{a,c} mils to [ ]^{a,c} mils at SPU conditions. From these calculations, it can be concluded that although there

may be an increase in the level of wear that would occur at the uprated operating conditions, the increased level is not significant.

In some models of steam generators, particular consideration is given to the potential for high cycle fatigue of U-bend tubes. This phenomenon has been observed in tubes with carbon steel support plates where denting or a fixed tube support condition has been observed in the uppermost plate. However, since the CPNPP Unit 2 steam generator tube support plates are manufactured from stainless steel, there is no potential for the necessary boundary conditions (that is, denting) to occur at the uppermost support plate. High cycle fatigue of U-bend tubes will not be an issue with the CPNPP Unit 2 Model D-5 steam generators.

With increased main feedwater flow (included in the uprate evaluation supporting the existing feed flow orifices) an increase in the rate of tube wear would be projected. However, this increase is anticipated to be modest and not cause rapid wear. With increased main feedwater flow, the following is anticipated:

- The rate of tube wear is projected to increase by a factor of about [ ]^{a,c} based on uprate design conditions and the maximum main feedwater flow. The projected level of tube wear would be expected to remain small however, because the current wear rate is relatively small and will not result in unacceptable wear for the general population of tubes or for those tubes exhibiting wear following eddy current testing.

Conclusion

The analysis of the CPNPP Unit 2 Model D-5 steam generators indicates that significant levels of tube vibration will not occur from either the fluid-elastic, vortex shedding, or turbulent mechanisms as a result of the 3,628 MWt NSSS SPU. In addition, the projected level of tube wear as a result of vibration would be expected to remain small because the current wear rate is relatively small and will not result in unacceptable wear for the general population of tubes or for those tubes exhibiting wear following eddy current testing.

Additional Steam Generator Hardware Evaluation

Mechanical repair hardware refers to components such as plugs and stabilizers that are installed in steam generator tubes to address tube degradation. These components were re-evaluated for the operating conditions and transients associated with the 3,628 MWt NSSS SPU. In anticipation of future tube plugging needs, both long and short 3/4-inch ribbed mechanical plugs were qualified for the SPU. In addition, since there are circumstances that may require tube ends to be reamed in association with plug removal, a remaining tube wall undercut thickness of [ ]^{a,c} inch was considered, and the resulting reduced weld joint geometry was qualified for continued service. Additionally, weld plugs, both those originally installed in the shop and those installed in the field, were addressed. The other hardware items that have been evaluated for the SPU are the Westinghouse collared stabilizers and the bare-cable stabilizers. All repair components evaluated were found acceptable.

Unit 2 Steam Generator Chemistry Evaluation

Introduction

Water chemistry of both the primary and secondary sides in nuclear power plants is controlled to maximize the long-term availability of PWR plants. Primary water chemistry control can, and has been, effectively used to control radiation field buildup on ex-core surfaces also. Guidelines have been provided to utilities by EPRI for primary and secondary chemistry (References 4 and 5). In addition, other organizations such as the Nuclear Energy Institute (NEI), have provided guidelines with respect to specific equipment (such as steam generators) that are incorporated into the EPRI guidelines. These documents form an industry consensus approach for chemistry programs that are embodied in and augmented by the plant-specific Strategic Water Chemistry Plans for the primary and secondary systems. Plant-specific strategies must consider plant design, ability to support chemistry control philosophies and targets, and cost-benefit tradeoffs. Uprates in power potentially affect water chemistry in the steam generator of the nuclear power plant because of changes in temperature and/or flow rates.

Input Parameters, Assumptions, and Acceptance Criteria

Input parameters include the operational parameters, EPRI primary and secondary chemistry guidelines (References 4 and 5), and strategic water chemistry plans for the primary and secondary sides at CPNPP.

A "Constant Elevated pH" chemistry program is followed for CPNPP Unit 2. This program was initiated on Unit 2 in Fuel Cycle 7 as a demonstration program and later implemented in CPNPP Unit 1 at Fuel Cycle 12. Morpholine and dimethylamine are used as the secondary amines in both units. In addition, hydrazine is maintained at three times the condensate dissolved oxygen and at a minimum of 20 ppb to address upset conditions. Historically, the secondary chemistry program followed at CPNPP has resulted in no identified steam generator tube degradation (for outside diameter stress corrosion cracking), as well as startup iron values that are the lowest reported values in the industry.

There are no acceptance criteria for this evaluation.

Description of Analyses and Evaluations

EPRI guidelines recognize the difference in design and operating characteristics of nuclear plants and prescribes that each plant generate strategic water chemistry plans for the primary and secondary water chemistries. This allows chemistry programs specifically tailored for each plant.

Results

CPNPP Unit 2 chemistry is controlled based upon strategies contained in the primary and secondary strategic chemistry programs.

Conclusion

No significant changes in the bulk chemistry of either the primary or secondary side are expected due to the 3,628 MWt NSSS SPU because the bulk chemistry will be continued to be controlled after the upgrade by plant procedures and specifications conforming to industry accepted guidelines and embodied in the CPNPP strategic water chemistry documents. In addition, the temperatures stated in uprated design parameters are in the range where other plants control bulk chemistry based on the same industry guidelines. Bulk water chemistry affects the chemistry in regions of the steam generators (or other components) where deleterious chemical species could accumulate and changes in temperatures and pressures can affect the superheat available and the amount of concentration possible in these regions. The potential for increased superheat in occluded environments underscores the importance of maintaining non-aggressive secondary side chemistry conditions.

2.2.2.5.2.3 References

1. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, 1971 plus addenda through Summer 1973.

2. NEI 97-06 Revision 2, "Steam Generator Program Guidelines," Nuclear Energy Institute, May 2005.

3. Technical Report 1012987, "Steam Generator Integrity Assessment Guidelines Revision 2," Electric Power Research Institute, July 2006.

4. EPRI PWR Primary Water Chemistry Guidelines, Volume 1, Revision 5, TR-105714-V1R5, March 2003.

5. EPRI Pressurized Water Reactor Secondary Water Chemistry Guidelines, Revision 6, 1008224, Final Report, December, 2004.

Table 2.2.2.5-8

CPNPP Unit 2 - Steam Generator Performance Characteristics at 3,425 and 3,475 MWt

Case	Original Case 3,425 MWt Power	A 3,475 MWt Power, 0% Plugging	B 3,475 MWt Power, 0% Plugging
SG T_{avg}, °F	589.06	592.53	592.53
Operating Conditions			
NSSS Power, MWt	3425	3475	3475
Power, MWt/SG	856.3	868.75	868.75
Primary Temperature			
SG T_{hot}, °F	618.82	621.85	621.85
SG T_{cold}, °F	559.30	563.20	563.20
SG T_{avg}, °F	589.06	592.53	592.53
Primary Flow, gpm	95,700	97,900	97,900
Feed Temperature, °F	440.0	444.6	390.0
Plugging, %	0	0	0
Operating Characteristics[1]			

a,c

Note:

1. The results shown here are applicable for zero to 100 gpm blowdown rate.

Table 2.2.2.5-9

CPNPP Unit 2 - Steam Generator Performance Characteristics with 3,628 MWt NSSS SPU

Case	**Ref. Case** **3,475 MWt Power**	**1A/1B** **3628 MWt Power, 0% Plugging**	**2A/2B** **3628 MWt Power, 10% Plugging**	**3A/3B** **3628 MWt Power, 0% Plugging, Elevated Inlet Temp**	**4A/4B** **3628 MWt Power, 10% Plugging, Elevated Inlet Temp**
SG T_{avg}, °F	592.50	574.10	574.10	589.00	589.00
Operating Conditions					
NSSS Power, MWt	3,475	3,628	3,628	3,628	3,628
Power, MWt/SG	868.8	907.0	907.0	907.0	907.0
Primary Temperature					
SG T_{hot}, °F	621.85	606.30	606.30	620.40	620.40
SG T_{cold}, °F	563.20	541.90	541.90	557.60	557.60
SG T_{avg}, °F[(1)]	592.50	574.10	574.10	589.00	589.00
Primary Flow, gpm	97,900	95,700	95,700	95,700	95,700
Feed Temperature, °F	444.6	450.3/390.0	450.3/390.0	450.3/390.0	450.3/390.0
Plugging, %	0	0	10	0	10
Operating Characteristics[(2)]					a,c

Notes:

1. In GENF, SG T_{avg} is the average of the steam generator primary inlet and outlet temperatures. The vessel T_{avg} is the average of the reactor vessel primary inlet and outlet fluid temperatures. The difference in the T_{avgs} is due to pump heat addition.
2. The results shown are applicable for 0 to 100 gpm blowdown rate.
3. PCWG steam pressures differ slightly from these values as a result of different codes used and different calculations for internal pressure drop.
4. The pressure drop values represent the differences in the pressures calculated at the feedwater nozzle inlet and at the steam nozzle outlet. They are based on 100 gpm blowdown rate.
5. Ratio of local quality at DNB on ATHOS runs. ATHOS analysis was performed only for the limiting case, that is, the case with 10% plugging and original inlet temperature.

Table 2.2.2.5-10

Unit 2 Uprate Evaluation Summary Primary Side Components

Component	Load Condition	Stress Category	Stress (ksi)/Fatigue		Allowable (ksi)/Fatigue	Comments
			Baseline	Uprate(4)		
Divider Plate	Normal/Upset	$\lvert P_m+P_b+Q \rvert$ – (Section 1, OS)	(1)	(1)	69.9	
	Fatigue	Fillet	[ ]^{a,c}	[ ]^{a,c}	1.0	(6)
		Fillet	[ ]^{a,c}	[ ]^{a,c}	1.0	(5)
		Drain Hole	[ ]^{a,c}	[ ]^{a,c}	1.0	(6)
		Drain Hole	[ ]^{a,c}	[ ]^{a,c}	1.0	(5)
Tubesheet	Normal/Upset	$\lvert P_m+P_b+Q \rvert$ - (TS Center-Upper Surface)(2)	[ ]^{a,c}	[ ]^{a,c}	122.0	Test allowable (2Sy)
	Fatigue	Tubesheet - Center-Upper Surface	[ ]^{a,c}	[ ]^{a,c}	1.0	(6)
		Tubesheet - Center-Upper Surface	[ ]^{a,c}	[ ]^{a,c}	1.0	(5)
	Normal/Upset	$\lvert P_m+P_b+Q \rvert$ - Upper Shell Junction - IS	[ ]^{a,c}	[ ]^{a,c}	122.0	
	Fatigue	Upper Shell Junction – IS	[ ]^{a,c}	[ ]^{a,c}	1.0	(6)
		Upper Shell Junction – IS	[ ]^{a,c}	[ ]^{a,c}	1.0	(5)
	Normal/Upset	$\lvert P_m+P_b+Q \rvert$ - Lower Shell Junction – "A" – IS	[ ]^{a,c}	[ ]^{a,c}	90.0	
	Fatigue	Lower Shell Junction – "A" – IS	[ ]^{a,c}	[ ]^{a,c}	1.0	(6)
		Lower Shell Junction – "A" – IS	[ ]^{a,c}	[ ]^{a,c}	1.0	(5)
Tube-to-Tubesheet Weld	Normal/Upset	$\lvert P_m+P_b+Q \rvert$ - Weld Root(3)	(1)	(1)	77.80	
	Fatigue	Weld Root – Section 1	[ ]^{a,c}	[ ]^{a,c}	1.00	(6)
		Weld Root – Section 1	[ ]^{a,c}	[ ]^{a,c}	1.00	(5)

Table 2.2.2.5-10 (cont.)

Unit 2 Uprate Evaluation Summary Primary Side Components

Component	Load Condition	Stress Category	Stress (ksi)/Fatigue		Allowable (ksi)/Fatigue	Comments
			Baseline	Uprate[4]		
Tubes	Normal/Upset	I P_m+P_b+Q I - Section B-B[3]	(3)	(3)	79.8	
	Fatigue	Section B-B	[ ]^{a,c}	[ ]^{a,c}	1.0	(6)
		Section B-B	[ ]^{a,c}	[ ]^{a,c}	1.0	(5)
Blowdown Pipe	Normal/Upset	I P_m+P_b+Q I - Weld Location - C	[ ]^{a,c}	[ ]^{a,c}	90.00	
	Fatigue	Radial Direction	[ ]^{a,c}	[ ]^{a,c}	1.00	(6)
		Radial Direction	[ ]^{a,c}	[ ]^{a,c}	1.00	(5)
		Hoop Direction	[ ]^{a,c}	[ ]^{a,c}	1.00	(6)
		Hoop Direction	[ ]^{a,c}	[ ]^{a,c}	1.00	(5)

Notes:

1. Exceeded $3S_m$ limit for primary-to-secondary stress intensity range. Per NB-3228.1 of the AMSE Code, the $3S_m$ limit can be exceeded if a plastic analysis is performed and shakedown is established. Fatigue analysis is performed based on the plastic analysis.
2. Based on Test Allowable -2Sy.
3. Exceeds $3S_m$. Simplified elastic-plastic analysis was done and Ke factors were used in fatigue calculation.
4. The uprate evaluation includes the effect due to the Low Temperature Overpressure Protection (LTOP) System. The baseline analysis does not include LTOP effect.
5. w LTOP
6. w/o LTOP

Table 2.2.2.5-11 Unit 2 Uprate Evaluation Summary Secondary Side Components						
Component	Load Condition	Stress Category	Stress (ksi)/Fatigue		Allowable (ksi/Fatigue	Comments
			Baseline	Uprate		
Main Feedwater Nozzle	Normal/Upset	$\vert P_m + P_b + Q \vert$ - Section G-G - OS	(1)	(1)	51.90	
	Fatigue	Section G-G - OS	[ ]a,c	[ ]a,c	1.0	FW Temp. = 390°F
		Section D-D - IS:	[ ]a,c	[ ]a,c	1.0	FW Temp. = 390°F
Auxiliary Feedwater Nozzle	Normal/Upset	Section A-A – IS	[ ]a,c	[ ]a,c	90.0	(Sections D-D, F-F, G-G, and H-H) (1)
	Fatigue	Section A-A – IS	[ ]a,c	[ ]a,c	1.0	
Secondary Manway Bolts	Normal/Upset	$\vert P_m + P_b + Q \vert$ - Bolt IS	[ ]a,c	[ ]a,c	85.8	
	Fatigue	Bolt (2)	[ ]a,c(2)	[ ]a,c(3)	1.0	
Steam Nozzle	Normal/Upset	$\vert P_m + P_b + Q \vert$ - Section A-A	[ ]a,c	[ ]a,c	90.0	
	Fatigue	Section A-A – IS	[ ]a,c	[ ]a,c	1.0	
	Normal/Upset	$\vert P_m + P_b + Q \vert$ - Insert Fillet Weld	(1)	(1)	78.0	
	Fatigue	Fillet Weld – Horizontal Section	[ ]a,c	[ ]a,c	1.0	Adjusted for pre-uprate operation
		Fillet Weld – Diagonal Section	[ ]a,c	[ ]a,c	1.0	
		Fillet Weld – Vertical Section	[ ]a,c	[ ]a,c	1.0	

Table 2.2.2.5-11 (cont.)

Unit 2 Uprate Evaluation Summary Secondary Side Components

Support Ring	Normal/Upset	$\mid P_m + P_b + Q \mid$ - Support Ring	(1)	(1)	56.10	
	Fatigue	Support ring - IS	[ ]^{a,c}	[ ]^{a,c}	1.0	
Wrapper Support System	Normal/Upset	$\mid P_m + P_b + Q \mid$	n/a	n/a		No impact from uprate
	Fatigue	All Support Components	[ ]^{a,c}	[ ]^{a,c}	1.0	

Notes:

1. Simplified elastic-plastic analysis was performed to show qualification per the requirements of the ASME Code.
2. Fatigue usage shown is for a 20-year replacement schedule.
3. Fatigue usage shown is for an 18-year replacement schedule.

Table 2.2.2.5-12 Bounding Parameters				
Program Description LR Section 1.1 Parameters	**SPU**			
	Low T_{avg}		**High T_{avg}**	
Feed Temperature, °F	390.0	450.3	390.0	450.3
NSSS Power, MWt	3,628	3,628	3,628	3,628
RCS Pressure, psia	2,250	2,250	2,250	2,250
Vessel Average Temperature, °F	574.2	574.2	589.2	589.2
Steam Temperature, °F	522.0	518.7	539.4	535.9
Steam Pressure, psia	826[1]	804[1]	958[1]	930[1]
Tube Plugging, %	10	10	10	10

Note:

1. 15 psi steam generator internal pressure drop incorporated

Table 2.2.2.5-13

Summary of ΔP for CPNPP Unit 2 3,628 MWt

a,c

2.2.2.5.3 Conclusion

Each of the preceding subsubsections that presented the evaluation results for Units 1 and 2, structural integrity, thermal-hydraulic performance, and tube vibration and wear concluded that the pertinent acceptance criteria are met. In addition, the steam generator supports were evaluated for the SPU conditions and found acceptable. Therefore, Luminant Power concludes that the CPNPP Units 1 and 2 steam generators are acceptable for operation at SPU conditions.

2.2.2.6 Reactor Coolant Pumps and Supports

2.2.2.6.1 Regulatory Evaluation

The reactor coolant pumps (RCPs) are described in FSAR Sections 3.9N, 5.1 and 5.4.1. The RCP supports are described in Final Safety Analysis Report (FSAR) Sections 3.9N and 5.4.14. Each reactor coolant loop (RCL) contains a vertical single-stage centrifugal type pump that employs a controlled leakage seal assembly. The Regulatory Evaluation included in Licensing Report (LR) subsection 2.2.2 also applies to the RCP and its supports.

The functions of the RCPs are:

- To maintain an adequate cooling flow rate by circulating a large volume of primary coolant water at high temperature and pressure through the reactor coolant system (RCS)
- To provide adequate flow coastdown to prevent core damage in the event of a simultaneous loss of power to all pumps
- To provide a portion of the reactor coolant pressure boundary (the pressure boundary parts of the RCP)

Current Licensing Basis

The generic current licensing basis in LR subsection 2.2.2 applies to the RCPs and supports, with the following amplifications.

The RCPs are single-speed centrifugal units driven by air-cooled, three-phase induction motors. The shaft is vertical with the motor mounted above the pumps. A flywheel on the shaft above the motor provides additional inertia to extend pump coastdown. The inlet is at the bottom of the pump; discharge is on the side. The RCPs employ a controlled leakage seal assembly. There are four RCPs per unit. FSAR Table 5.4-1 provides RCP design parameters.

FSAR Table 5.2-2 indicates that the internals portion of the RCP, which contacts or may contact primary system fluid, including forgings, castings, tube and pipe, pressure plates, bars, and closure bolting, are made from stainless steel.

FSAR Section 3.2 and Table 17A-1 provide Seismic Classification, Safety Classification, and applicable American Society of Mechanical Engineers (ASME) Code Category for structures, systems, and components (SSCs) such as the RCPs. FSAR Table 17A-1 shows that certain parts of the RCP (RCP casing, main flange, thermal barrier, seal housing and pressure retaining bolting) are classified as ASME Section III, Class 1, American Nuclear Society (ANS) Safety Class 1. The balance of the RCP components (motors, seal housings, etc.) are classified as ANS Safety Class 2. RCP supports are designed to meet the same Safety Class designation as the components they support. The RCP supports are Safety Class 1.

The RCP and supports are designed to withstand stresses originating from various operating design transients. FSAR Table 3.9N-1 summarizes RCS design transients for normal, upset, emergency, faulted, and test conditions. It indicates that the RCS and the RCP are designed for 200 heat up transients of 100°F per hour, and an additional 200 cooldown transients of 100°F per hour.

2.2.2.6.2 Technical Evaluation

The RCP is tested and inspected per the requirements of Section XI of the ASME Boiler and Pressure Vessel (B&PV) Code and designed to ASME Code Section III, 1971 Edition through Summer 1973. The RCP flywheels are subject to an ISI once every 20 years.

Input Parameters, Assumptions, and Acceptance Criteria

The major inputs used in the RCP evaluation are the NSSS parameters provided in LR Section 1.1, and the NSSS design transients provided in LR subsection 2.2.6. These sections provide the operating and transient conditions for the SPU conditions. The RCPs are installed in the RCS cold leg, between the steam generator outlet and the reactor vessel inlet. Therefore, the cold leg temperatures and the cold leg transients are applicable to the RCPs. These operating and transient conditions differ in some cases from those specified in the RCP equipment specification, to which the CPNPP Units 1 and 2 RCPs were already designed and analyzed.

The SPU parameters (LR Section 1.1) and the NSSS system design transient parameters (LR subsection 2.2.6) were considered in the SPU evaluations. These two sections contain all of the pressure or thermal-hydraulic design parameters due to the SPU that would affect the RCPs or their supports. Design loads under SPU conditions were found to be less than or equal in magnitude to the loads that were previously analyzed, with no changes to the load application points or number of occurrences.

The inputs for seismic analysis of the RCP, including seismic accelerations and pump component mass and stiffness, have not changed due to the SPU conditions. The power required to operate the pump under the SPU conditions remains within the capability of the motor. Therefore, hardware changes are not required and seismic analyses and non-pressure boundary component evaluations are unaffected by the SPU.

The evaluation of the RCPs for the SPU compared the operating temperatures and pressures defined in the SPU parameters to the pressures and temperatures considered in previous analyses of the RCPs. In addition, the NSSS design transients for the SPU were compared to the transients considered in previous evaluations. Where temperatures, pressures, and NSSS transients considered in previous analyses enveloped the temperatures, pressures, and NSSS transients defined for the SPU, no additional analysis was required. For the inputs that were not enveloped by the previously analyzed parameters, RCP structural analyses and evaluations were performed as necessary to incorporate the revised design inputs. Where these analyses and evaluations were required, the acceptance criteria were that the Units 1 and 2 RCP

pressure boundary components meet the stress limits and fatigue usage requirements of the ASME Code, Section III for plant operation with the SPU conditions.

The RCP motors were evaluated for the NSSS parameters and best-estimate flows provided in LR Section 1.1 at an assumed NSSS power of 3,628 MWt. The input parameters considered in the evaluation of the RCP motors for the Units 1 and 2 SPU Program are for a range of steam generator outlet temperatures from 541.9° to 557.6°F. The range of best-estimate flows considered is from 101,900 to 100,200 gpm/loop (Unit 1) and 100,400 to 98,300 gpm/loop (Unit 2) for a range of steam generator tube plugging (SGTP) from 0- to 10-percent SGTP at full-power operation. For the cold condition (70°F), the range of best-estimate flows considered is from 96,000 to 94,000 gpm/loop (Unit 1) and 94,800 to 92,600 gpm/loop (Unit 2) for 0- to 10-percent SGTP.

The steam generator outlet (RCP inlet) temperatures and best-estimate flows were considered in a hydraulic analysis using the operating characteristics of the RCPs. This hydraulic analysis calculates the power requirements for the impeller that operates at the highest power for both hot and cold operation. The RCP motors were evaluated to confirm that they continue to meet their design requirements.

The RCL piping loads on the RCP supports due to deadweight, thermal expansion, seismic operating basis earthquake (OBE), and seismic safe shutdown earthquake (SSE) loading cases are obtained from the evaluation for the RCL piping system analyses for the SPU program as described in LR subsection 2.2.2.1, NSSS Piping, Component, and Supports. The RCP supports have been evaluated and remain acceptable.

Description of Analyses and Evaluations

Operating Temperature and Pressure

The SPU parameters were used to evaluate the acceptability of the RCPs. In the SPU parameters, there are no changes from the current reactor coolant pressure of 2,250 psia for any of the SPU cases. For the SPU, the RCS cold leg temperature (T_{cold}), defined by the vessel inlet (RCP outlet) temperature, has a maximum of 558.0°F and a minimum of 542.2°F. The equipment specification for both units specifies an operating temperature of 558.6°F. Since lower temperatures result in higher allowable stresses for the pressure boundary materials and since the SPU NSSS parameters are bounded by those defined in the equipment specification, no further evaluation of the RCP pressure boundary integrity was required for the operating temperature and pressure associated with the SPU.

Transient Discussion

The NSSS design transients were recalculated for the SPU program and are provided in LR subsection 2.2.6, NSSS Design Transients. The cold leg transients were applicable to the RCP evaluation.

For Unit 1, the current cold leg design transients have not changed. Because all of the Unit 1 components previously passed the fatigue requirement, and the Unit 1 cold leg design transients have not changed from that evaluation, the Unit 1 components continue to meet the fatigue requirement.

For Unit 2, the recalculated transients had some temperature and pressure changes that were different than those in the design transients given in the equipment specification or used in the original analyses. The components that did not meet fatigue waiver requirements, and for which fatigue evaluations were previously required, are now considered.

The Unit 2 evaluations follow.

Unit 2 RCP Main Closure Bolted Joint Stress Analysis

This analysis showed that the main transients with a potential for affecting the fatigue usage of the main closure bolted joint are the heatup and cooldown transients and the newly defined Low-Temperature Overpressure Protection (LTOP) transient. The heatup and cooldown transients remained unchanged for the SPU condition. The number of pressure cycles associated with the LTOP transient impacted the fatigue waiver of the thermowell, the thermal barrier flange, and main flange bolts. Therefore, the stress ranges and the cumulative usage factors for the thermowell, thermal barrier flange, and the main flange bolts were recalculated. The stresses remain within the ASME Code allowable values. The values of the stresses and the cumulative usage factors are shown in Tables 2.2.2.6-1 and 2.2.2.6-2.

Unit 2 Main Closure Stress Analysis

The main closure consists primarily of the main flange, thermal barrier flange, and main flange bolts. The main flange is protected from the high reactor coolant temperatures by the thermal barrier, and thus sees much lower temperatures than those in the parts of the pump directly exposed to the primary coolant. This protection also means that the effect of the primary system cold leg transients on the main flange is small, except for the heatup and cooldown transients. The addition of the new transients and revised temperature ranges only impacted the fatigue waiver calculation for the thermowell, thermal barrier flange, and the main flange bolts. The recalculated cumulative usage factors and the stresses remain within the ASME Code allowable values. The cumulative usage factors and stresses are shown in Tables 2.2.2.6-1 and 2.2.2.6-2.

Unit 2 Pump Casing Stress Analysis

In most cases, the CPNPP Units 1 and 2 SPU transients were very similar to the transients considered in the original analysis. The main exception to this was the addition of the LTOP transient's event. The LTOP event did not exist in the original analyses. The LTOP temperature transient has been shown to induce general casing section temperature range changes bounded by the original analysis temperature transients. Similarly, the range of LTOP pressure transients is bounded by original analysis pressure transients. Therefore, LTOP did not change the existing temperature and pressure stress ranges and the effect of the LTOP transient was

assessed by simply evaluating the changes to the cumulative usage from the additional 10 thermal cycles and 6,000 pressure cycles. The stress intensities and cumulative usage remain less than the ASME Code allowable values. The casing nozzles exceeded the $3S_m$ requirement, thus necessitating a simplified elastic-plastic analysis. Fatigue usage factors were calculated as part of the simplified elastic-plastic analysis and were less than the limit of one. Thus, both the suction and discharge casing nozzles continue to meet the fatigue requirement. The usage factors and the stresses are shown in Tables 2.2.2.6-1 and 2.2.2.6-2.

Support Foot Analysis

The support foot was considered a structural member in the original stress analysis and was analyzed only for mechanical loads. There was no transient analysis. Thus, changes to the NSSS design transients associated with the SPU did not affect the support foot analysis.

Flow-Induced Vibration Analysis

Analysis of flow-induced vibration is not included in the licensing basis for CPNPP Units 1 and 2. The change in RCS flow under SPU conditions is not significant considering the heavy construction of the RCPs.

RCP Motors

For the RCP motors, a hydraulic analysis was performed using best-estimate flows and modeling the characteristics of the RCPs. The hydraulic analysis is used to calculate brake horsepower for the RCP motors, the loading on the thrust bearings, and the torque-speed curve for the RCP motors.

The RCP motors were evaluated in the following four areas for the SPU conditions:

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

The RCP motor brake horsepower results from the hydraulic analysis are as given in Table 2.2.2.6-3, Reactor Coolant Pump Motor Performance Summary. The worst-case hot loop load under the SPU operating conditions is 7,226 hp. The worst-case cold loop load under the SPU operating conditions is 9,245 hp. These loadings are more than the motor nameplate ratings of 7,000 hp for hot loop operation and 8,750 hp for cold loop operation. This necessitates the need to calculate the predicted stator winding temperature rise value for both the hot loop and cold loop conditions.

Per the equipment specification, the motor is required to drive the pump continuously under hot loop conditions without exceeding a stator winding temperature rise of 75°C (corresponding to the National Electrical Manufacturer's Association (NEMA) MG-1, Section III, Part 20.8, Class B temperature rise limit in a 50°C ambient). The RCP motors are acceptable for hot loop

operation at the increased steady-state power output level because the calculated stator winding temperature rise value does not exceed that allowed by the equipment specification. The predicted temperature rise will be 68.2°C, whereas the Class B insulation rating allows a stator temperature rise of 75°C above the 50°C ambient. Therefore, continuous operation at the revised hot loop loading is acceptable.

Per the equipment specification, the motor is required to drive the pump for up to 50 hours (continuous) and 3,000 hours maximum over the 40-year design life under cold loop conditions without exceeding a stator winding temperature rise of 100°C (corresponding to the NEMA MG-1, Section III, Part 20.8, Class F temperature rise limit in a 50°C ambient). Analysis indicates that the cold loop stator winding temperature rise at the revised load of 9,245 HP will be approximately 106°C, which is in excess of the NEMA limit. Further analysis predicts that exceeding the Class F limit by 6°C during a maximum of 3,000 hours of operation will accelerate the thermal aging of the insulation and reduce the life of the motor by approximately 2 months from the 40-year design life, or 0.4 percent. This evaluation concludes that the RCP motors under cold loop loading remain acceptable due to the insignificant impact on expected motor life and the conservative methodology used in the analysis.

Per the equipment specification, the motor is required to start across the line with a minimum 80-percent starting voltage against the reverse flow of the pumps running at full speed, under cold loop conditions. The limiting component for this type of starting duty is the rotor cage winding, which has design limits per the equipment specification of a 300°C temperature rise on the rotor bars and a 50°C temperature rise on the rotor resistance rings. Using the torque-speed curve from the hydraulic analysis, a conservative all-heat-stored analysis showed a bar temperature rise of 254.8°C and a resistance ring temperature rise of 28.71°C, both of which are within their allowable limit. Therefore, the motor can safely accelerate the load under worst-case conditions.

The axial down thrust for the SPU conditions decreases from 55,000 to 52,025 pounds for hot reactor coolant operation and decreases from 75,000 to 72,231 pounds for cold reactor coolant operation. For the hot loop condition, the decrease in the down thrust loads actually increases the net up thrust on the bearings by 2.9 percent. For the cold loop condition, the decrease in the down thrust decreases the net down thrust on the bearings. The thrust bearing is designed for loads exceeding 101,200 pounds. Therefore, these changes in thrust loads are not significant. There will be no impact on thrust bearing performance due to the revised loads.

There are no changes required as a result of the SPU for the RCP and RCP motor supporting systems such as cooling water, seal injection flow, or lubricating oil.

2.2.2.6.3 Conclusions

The evaluations related to the structural integrity of the RCP and supports have adequately addressed the effects of the proposed SPU on the RCPs, RCP motors, and RCP supports. The evaluations have demonstrated that the RCPs, RCP motors, and supports continue to meet the CPNPP Units 1 and 2 current licensing basis requirements with respect to General Design Criterion (GDC)-1, -2, -4, -14 and -15 following implementation of the SPU. Therefore, Luminant Power finds the SPU acceptable with respect to the design of the RCPs, RCP motors, and supports.

Table 2.2.2.6-1 Summary of Fatigue Results for CPNPP RCP Pressure Boundary Components for the SPU				
Unit and Component	**Fatigue Stress Amplitude, psi**		**Fatigue Usage**[1]	
	Previous	**Current**	**Previous**	**Current**
All Unit 1 Components	Continue to meet fatigue waiver or fatigue usage requirements since applicable transients unchanged			
Casing, Main Flange, Seal Housing, and Weir Plate (Unit 2)	Not applicable	Not applicable	Meets fatigue waiver	Meets fatigue waiver
Thermowell (Unit 2)	29,968	31,470	0.53	0.72
Main Flange Bolts (Unit 2)	167,320	175,592	0.8	0.995
Thermal Barrier Flange (Unit 2)	26,582	26,582	0.0002	0.0002
Casing Suction Nozzle (Unit 2)	Not applicable	Not applicable	Meets fatigue waiver	Meets fatigue waiver
Casing Discharge Nozzle (Unit 2)	35,828	41,919	0.0005	0.001
Casing to Discharge Nozzle Intersection (Unit 2)	229,000	256,046	0.209	0.729
Casing Large Feet (Unit 2)	82,873	82,873	0.083	0.083
Auxiliary Nozzle (Unit 2)	255,398[2]	255,398[2]	0.93	0.93
Thermal Barrier Heat Exchanger (Unit 2)	221,193[2]	221,193[2]	0.31	0.31

Notes:

1. Fatigue requirement is that the fatigue waiver be met or the fatigue usage is less than one.
2. Stress amplitude for one time application of worst-case cycle.

Table 2.2.2.6-2 Summary of Stress Intensity Range Results for CPNPP RCP Pressure Boundary Components for the SPU			
Unit and Component	**Previous Stress Intensity Range, psi**	**Current Stress Intensity Range, psi**	**Allowable Stress Intensity Range, $3S_m$, psi**
All Unit 1 Components	Continue to meet $3S_m$ or alternative simplified elastic-plastic analysis requirement since applicable transients unchanged		
Casing, Main Flange, Thermal Barrier Flange, Seal Housing Assembly, Casing Feet (Unit 2)	Continue to meet $3S_m$ or alternative simplified elastic-plastic analysis requirement since applicable transients unchanged		
Thermowell (Unit 2)	41,529	43,605	60,000
Thermal Barrier Heat Exchanger (Unit 2)	30,280	30,280	60,000
Weir Plate (Unit 2)	43,280	45,012	48,000
Casing Suction Nozzle (Unit 2)	46,065	45,772	53,250
Casing Discharge Nozzle (Unit 2)	34,991 (54,285 before simplified elastic-plastic analysis)	35,978 (55,272 before simplified elastic-plastic analysis)	49,590
Casing to Discharge Nozzle Intersection (Unit 2)	40,719 (98,075 before simplified elastic-plastic analysis)	40,719 (109,844 before simplified elastic-plastic analysis)	49,590

Table 2.2.2.6-3

Reactor Coolant Pump Motor Performance Summary

Design Parameter	Current Condition or Rating	Uprating Case	Change or Margin
Hot Loop Load	7000 hp (Nameplate Rating)	7226 hp	+226 hp
Cold Loop Load	8750 hp (Nameplate Rating)	9245 hp	+495 hp
Hot Loop Stator Temperature Rise	66.6°C (by test) (75°C NEMA limit)	68.2°C	+1.6°C (6.8°C margin)
Cold Loop Stator Temperature Rise	99.3°C (estimated) (100°C NEMA limit)	106.0°C	+6.7°C (-6.0°C margin)
Starting Rotor Bar Temperature Rise	300°C (Design Limit)	254.8°C	45.2°C margin
Starting Rotor Resistance Ring Temperature Rise	50°C (Design Limit)	28.71°C	21.29°C margin
Axial Thrust (Hot Loop)	-55,000 lbs (Design Condition)	-52,025 lbs	2,975 lb increase
Axial Thrust (Cold Loop)	-75,000 lbs (Design Condition)	-72,231 lbs	2,769 lb increase

2.2.2.7 Pressurizer and Supports

2.2.2.7.1 Regulatory Evaluation

The pressurizer and supports are reviewed as part of the SPU. The Comanche Peak Nuclear Power Plant (CPNPP) Final Safety Analysis Report (FSAR) Section 5.4.10 describes the pressurizer and FSAR Section 5.4.14.2.4 describes the pressurizer support system.

FSAR Table 17A-1 provides Seismic Classification, Safety Class, Quality Assurance (QA), and applicable American Society of Mechanical Engineers (ASME) Code Category for Structures, Systems, and Components (SSCs) such as the pressurizer. FSAR Table 17A-1 states in part that, with the exception of its heaters, the pressurizer is classified as ASME Section III, Class 1, American Nuclear Society (ANS) Safety Class 1. The pressurizer supports are designed to meet the same Safety Class designation as the component they support. The pressurizer supports are Safety Class 1. FSAR Table 5.4-9 provides pressurizer design parameters.

Spray line nozzle, relief valve, and safety valve connections are located in the tip head of the vessel. Spray flow is modulated by automatically controlled air-operated valves. The spray valves also can be operated manually by a switch in the Control Room.

A small, continuous spray flow is provided through a manual bypass valve around the spray valves to assure that the pressurizer liquid is homogeneous with the coolant and to prevent excessive cooling of the spray piping. The bypass line provides a tempering effect on the spray nozzle and piping. The effects of the bypass line are not considered in the stress analysis of the pressurizer.

During an outsurge from the pressurizer, flashing of water to steam and generating of steam by automatic actuation of the heaters keep the pressure above the minimum allowable limit. During an insurge from the reactor coolant system (RCS), the spray system, which is fed from two cold legs, condenses steam in the vessel to prevent the pressurizer pressure from reaching the setpoint of the power-operated relief valves for normal design transients. Heaters are energized on high water level during insurge to heat the subcooled surge water that enters the pressurizer from the reactor coolant loop.

Material specifications are provided in FSAR Table 5.2-2 for the pressurizer, pressurizer relief tank, and the surge line. Additional details on the pressurizer design cycle analysis are given in FSAR Section 5.4.10.3.5.

The supports for the pressurizer are designed as follows:

1. A steel ring plate between the pressurizer skirt and the supporting concrete slab. The ring serves as a leveling and adjusting member of the pressurizer and may also be used as a template for positioning the concrete anchor bolts.

The upper lateral support consists of struts cantilevered off the compartment walls that bear against the "seismic lugs" provided on the pressurizer. The configuration of the lateral struts

depends on the location of the concrete walls and piping within the compartment, as well as the orientation of the pressurizer.

2.2.2.7.2 Technical Evaluation

The pressurizer is a vertical, cylindrical vessel with hemispherical top and bottom heads constructed of carbon steel, with austenitic stainless steel cladding on all surfaces exposed to the reactor coolant. A stainless steel liner or tube may be used in lieu of cladding in some nozzles.

The surge line nozzle and removable electric heaters are installed in the bottom head. The heaters are removable for maintenance or replacement. A thermal sleeve is provided to minimize stresses in the surge line nozzle. A screen at the surge line nozzle and baffles in the lower section of the pressurizer prevent an insurge of cold water from flowing directly to the steam/water interface and assist mixing.

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

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

The limiting operating conditions of the pressurizer occur when the RCS pressure is high and the RCS hot leg (T_{hot}) and cold leg (T_{cold}) temperatures are low. This maximizes the ΔT that is experienced by the pressurizer. Due to flow out of and into the pressurizer during various transients, the surge nozzle alternately sees water at the pressurizer temperature (T_{sat}) and water from the RCS hot leg at T_{hot}. If the RCS pressure is high (which means, correspondingly, that T_{sat} is high) and T_{hot} is low, then the surge nozzle will see maximum thermal gradients; and, therefore, experience the maximum thermal stress. Likewise, the spray nozzle and upper shell temperatures alternate between steam at T_{sat} and spray water, which, for many transients, is at T_{cold}. Therefore, if RCS pressure is high (T_{sat} is high) and T_{cold} is low, then the spray nozzle and upper shell will also experience the maximum thermal gradients and thermal stresses.

An evaluation was performed to support the CPNPP SPU Program to address the impact on the pressurizer. This evaluation is based on the range of NSSS operating parameters to support a nuclear steam supply system (NSSS) power level of 3,628 MWt.

Weld overlays have been performed on Unit 1 pressurizer nozzles and will be performed on Unit 2 prior to implementation of the uprate. The weld overlays were shown to have no significant effect on stress and fatigue results of existing analyses performed per Section III of the ASME Code. It was concluded that the current Section III analysis of record remains applicable for the surge, spray, and safety and relief nozzles.

Input Parameters, Assumptions, and Acceptance Criteria

Input Parameters

The reactor vessel outlet (T_{hot}) and reactor vessel inlet (T_{cold}) temperatures from the Performance Capability Working Group (PCWG) parameters in LR Section 1.1 define the normal operating temperatures for the surge and spray lines to the pressurizer. The reactor coolant pressure from LR Section 1.1 defines the pressurizer normal operating pressure (NOP) (2,250 psia) and saturated temperature (652.7°F). The minimum values of T_{hot} and T_{cold} from all cases in LR Section 1.1 are used in the evaluation of the pressurizer.

The NSSS design transients for Unit 1 were recently developed for the Δ76 SG in conjunction with a SPU to 3,582 MWt. These were reviewed for their continued applicability at a higher NSSS full power of 3,628 MWt for Unit 1. They were also reviewed to determine if they could also be applicable to Unit 2 with the Model D-5 steam generators.

A majority of the revised primary side transient parameters (hot and cold leg temperatures, RCS pressure and flow) for the CPNPP SPU Program (Licensing Report (LR) subsection 2.2.6) remain applicable for the CPNPP SPU Program. There are some exceptions in T_{hot} and T_{cold} for Unit 2, however, due to the steam generator model differences and the implications the generator has on primary side transients. The pressurizer parameters (pressure, temperature, surge flow) remain applicable for both units.

The pressurizer temperature and pressure variations, surge line flow rates, reactor vessel T_{hot} and T_{cold} for the NSSS design transients in LR subsection 2.2.6 are applicable to the pressurizer.

The uprate parameters provided in LR Section 1.1 and the NSSS uprate design transients given in LR subsection 2.2.6 provide the operating and transient conditions for those operating and transient conditions that differ from those previously addressed by which the CPNPP pressurizers were already designed and analyzed.

The design inputs are:

- Design pressure: 2,485 psig
- Design temperature: 680°F
- Normal operating pressure: 2,235 psig
- Normal operating temperatures:
  - T_{cold} temperature: 542.2° to 558.0°F
  - T_{hot} temperature: 606.2° to 620.4°F
- Zero-load temperature: 557°F

Assumptions

The PCWG uprate parameters and NSSS design transient uprate parameters are considered in the uprate evaluations. No other changes are considered to the pressure or thermal-hydraulic design parameters for the CPNPP SPU Program.

Unless indicated otherwise, the transients are assumed to be initiated with the pressurizer at the normal conditions for power operations, that is, saturation at 2,250 psia. The water and steam volumes are assumed to be saturated liquid and saturated vapor, respectively, and the temperature is 652.7°F.

Where pressurizer water temperature and/or steam temperature curves are not provided, these parameters are assumed to be the saturation temperature for the existing pressurizer pressure.

The relatively stagnant water normally in the spray piping is swept through the piping and into the pressurizer ahead of the spray flow from the cold leg. This water is assumed to be at 530°F. After the spray piping is swept out, the spray temperature is the same as the cold leg temperature.

Step temperature changes are assumed for components in contact with the spray and surge line insurges.

Seismic analyses and non-pressure boundary component evaluations are considered to be unaffected by the CPNPP SPU Program.

Acceptance Criteria

The initial set of acceptance criteria for evaluating design inputs affecting the pressurizer stress reports by comparison with the design inputs considered in LR Sections 1.1 and 2.2.6 are as follows:

Hot and cold leg temperatures remain within the ranges of the operating temperatures that have previously been considered and justified in the pressurizer stress reports.

NSSS design transients are less than or equal to the design transients previously considered in the pressurizer stress reports with regard to both severity and numbers of occurrences. Additionally, no new NSSS design transients that have not previously been considered are identified. The pressurizer temperature and pressure variations for each transient were considered in this comparison review to determine the relative severity of the revised design transients compared to the existing design transients.

Design loads are less than or equal in magnitude to the loads that were previously considered in the pressurizer stress reports with no changes to the load application points and numbers of occurrences.

If comparison of the design inputs for the CPNPP SPU Program with the current design inputs reveal hot and/or cold leg temperatures, NSSS design transients, or design loads that do not comply with the above criteria, then pressurizer structural analyses and evaluations must be performed, as necessary, to incorporate the revised design inputs. The acceptance criterion is that the CPNPP Units 1 and 2 pressurizer components meet the stress/fatigue analysis requirements of the ASME Code, Section III (Reference 1) for the plant operation in accordance with the CPNPP SPU Program.

Description of Analyses and Evaluations

The reactor vessel outlet (T_{hot}) and the reactor vessel/core inlet (T_{cold}) temperatures from LR Section 1.1 define the normal operating temperatures for the surge and spray lines to the pressurizer. The reactor coolant pressure defines the pressurizer NOP (2,250 psia) and saturated temperature (652.7°F). The minimum values of T_{hot} and T_{cold} from all cases in LR Section 1.1 are used in the evaluation of the pressurizer. The majority of the NSSS design transients did not change and those that did were enveloped by the existing design transients.

For the components at an NOP of 2250 psia, affected by T_{hot} (such as the surge nozzle), the temperature difference of 46.5°F for the revised parameter is bounded by the original equipment specification. The same is true for those components at an NOP of 2,250 psia, affected by T_{cold} (such as the spray nozzle), the temperature difference of 110.5°F for the revised parameter is bounded by the original equipment specification as well.

Parameter	ΔT (°F) Original Equip. Spec.	ΔT (°F) LR Section 1.1
T_{hot}	110	46.5
T_{cold}	125	110.5

The input parameters associated with the CPNPP SPU Program were reviewed and compared to the design inputs considered in the current pressurizer stress reports. The majority of the NSSS design transients did not change, and those that did were enveloped by the existing design transients.

Pressure fluctuations during the revised transients are the same or enveloped by the pressures in the original evaluations. It should be noted that the maximum pressure within each load category (Normal, Upset, Emergency, Faulted, and Test) has not changed from the value used in the original evaluations. Therefore, the revised transients have no effect on the primary stress evaluations performed previously.

The ΔTs between the pressurizer and the incoming T_{hot} and T_{cold}, as well as the variation in the pressurizer steam temperature, were determined for each of the Normal and Upset transients that were changed. As shown in Table 2.2.2.7-1, the revised ΔTs were less than those used in the original evaluations for all cases.

Results

The analysis performed here shows that the CPNPP SPU Program transients will have a limited effect on the pressurizer components. Design, Emergency, Faulted, and Test condition stresses remain unchanged. The maximum primary-plus-secondary stress intensity ranges and fatigue usages for Normal and Upset conditions also remain unchanged, as shown in Tables 2.2.2.7-2 and 2.2.2.7-3.

In addition, the supports were reviewed and there is no impact on the current supports.

2.2.2.7.3 Conclusions

All critical components of the CPNPP Units 1 and 2 pressurizers were evaluated for operation at the uprated conditions. It was determined that all ASME Code stress limits remain satisfied for all components, for all proposed operating conditions. In addition, the supports have been evaluated and there is no impact on the current supports.

2.2.2.7.4 Reference

1. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III, 1971 Edition with Addenda through Summer 1973.

Table 2.2.2.7-1 Maximum ΔTs of CPNPP Units 1 and 2 Revised Transients				
	$(T_{PZR} - T_{hot})^{(1)}$		$(T_{PZR} - T_{cold})^{(2)}$	
Transient	**LR Section 2.2.6**	**Stress Report**	**LR Section 2.2.6**	**Stress Report**
Loss of Load	60.9	75	143	150
Loss of Power(3)	60	80	0	0
Reactor Trip – B(3)	96	150	0	0
Inadvertent RCS Depressurization	96.5(4)	125	621	625
Control Rod Drop(3)	89.7	125	0	0
Inadvertent Safety Injection	95.3	125	129	135

Notes:

All temperatures are °F.

1. During insurges.
2. During sprays.
3. No spray during this transient.
4. At 60 seconds, where T_{hot} has dropped 50°F and the water in the pressurizer has not yet responded to the drop in pressure (conservatively assumed to still be at 652.7°F).

Table 2.2.2.7-2 CPNPP Units 1 and 2 Primary-Plus-Secondary Stress Intensity Ranges	
Component	***Calculated/Allowable***(1)
Surge Nozzle	[ ]c,e
Spray Nozzle	[ ]c,e
Safety and Relief Nozzles	[ ]c,e
Lower Head – Heater Penetrations	[ ]c,e
Heater Well	[ ]c,e
Upper Head and Shell	[ ]c,e
Support Skirt – Near Lower Head	[ ]c,e
Support Skirt – at Flange	[ ]c,e
Seismic Support Lug	[ ]c,e
Shell at Seismic Support Lug	[ ]c,e
Manway	[ ]c,e
Manway Bolt	[ ]c,e
Instrument Nozzle	[ ]c,e
Immersion Heater	[ ]c,e
Valve Support Bracket	[ ]c,e
Trunnion Buildup	[ ]c,e

Notes:

1. Ratio of calculated to allowable stress intensity.
2. The $3S_m$ limit on the range of primary-plus-secondary stress intensity may be exceeded provided the rules of NB-3228.3 of the ASME Code are met. Those requirements have been satisfied for this component.

Table 2.2.2.7-3 CPNPP Units 1 and 2 Fatigue Usages	
Component	**Fatigue Usage**
Surge Nozzle	[ ]c,e
Spray Nozzle	[ ]c,e
Safety and Relief Nozzles	[ ]c,e
Lower Head – Heater Penetrations	[ ]c,e
Heater Well	[ ]c,e
Upper Head and Shell	[ ]c,e
Support Skirt – Near Lower Head	[ ]c,e
Support Skirt – at Flange	[ ]c,e
Seismic Support Lug	[ ]c,e
Shell at Seismic Support Lug	[ ]c,e
Manway	[ ]c,e
Manway Bolt	[ ]c,e
Instrument Nozzle	[ ]c,e
Immersion Heater	[ ]c,e
Valve Support Bracket	[ ]c,e
Trunnion Buildup	[ ]c,e

2.2.3 Reactor Pressure Vessel Internals and Core Supports

2.2.3.1 Regulatory Evaluation

Reactor pressure vessel (RPV) internals consist of all the structural and mechanical elements inside the reactor vessel, including core support structures. Luminant Power reviewed the effects of the proposed stretch power uprate (SPU) on the design input parameters and the design basis loads and load combinations for the reactor internals for normal operation, upset, emergency, and faulted conditions. These include pressure differences and thermal effects for normal operation, transient pressure loads associated with loss-of-coolant accidents (LOCAs), and the identification of design transient occurrences. The Luminant Power review covered the analyses of flow-induced vibration (FIV) for safety-related and non-safety-related reactor internal components, as well as the analytical methodologies, assumptions, American Society of Mechanical Engineers (ASME) Code editions, and computer programs used for these analyses. The Luminant Power review also included a comparison of the resulting stresses and cumulative usage factor (CUF) against the corresponding Code-allowable limits. The acceptance criteria for this review are:

- 10 CFR 50.55a and General Design Criterion (GDC)-1, insofar as they require that SSCs important to safety be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed.

- GDC-2, insofar as it requires that structures, systems, and components (SSCs) important to safety be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions.

- GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.

- GDC-10, insofar as it requires that the reactor core be designed with appropriate margin to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the adequacy of CPNPP Units 1 and 2 design relative to:

- GDC-1 is described in FSAR Section 3.1.1.1, General Design Criterion 1 – Quality Standards and Records.

  SSCs important to safety are designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.

  Quality standards applicable to safety-related SSCs are generally contained in codes such as the ASME Boiler and Pressure Vessel Code. The applicability of these codes is specifically identified throughout this report and is summarized in FSAR Section 3.2.5. FSAR Chapter 17 provides direct reference to the Quality Assurance (QA) Program established to provide assurance that safety related SSCs satisfactorily perform their intended safety functions. The procedures for generating and maintaining appropriate design, fabrication, erection, and testing records are contained within the referenced documents.

- GDC-2 is described in the FSAR Section 3.1.1.2, General Design Criterion 2 – Design Bases for Protection Against Natural Phenomena.

  Features of the facility essential to accident prevention and mitigation of accident consequences, which are designed to withstand the effects of natural phenomena, are:

  1. The reactor coolant pressure boundary and containment barriers

  2. The controls and emergency cooling systems whose functions are to maintain the integrity of these barriers

  3. Reactivity systems, monitoring systems, and fuel systems

  All piping, components, and supporting structures of the reactor and safety-related systems are designed to withstand a specified seismic disturbance and credible combinations of effects of normal and accident conditions coincident with the effects of natural phenomena. Plant design criteria specify that there is to be no loss of function of such equipment in the event of the safe shutdown earthquake (SSE) ground acceleration acting in the horizontal and vertical directions simultaneously. The dynamic response of Seismic Category I structures to ground acceleration, based on an envelope of characteristics of the site foundation soils and on the critical damping of the foundation and structures, is included in the design analysis.

  Unit design criteria that ensure protection against natural phenomena are described in FSAR Section 3.2 (Classification of SSCs), Section 3.3 (Wind and Tornado Loadings), Section 3.4 (Water Level Design), and Section 3.7 (Seismic Design).

- GDC-4 is described in the CPNPP Units 1 and 2 FSAR Section 3.1.1.4, Environmental and Dynamic Effects Design Bases.

  SSCs important to safety are designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operating, maintenance, testing, and postulated accidents including LOCAs. These items are either protected from accident conditions or designed to withstand, without failure, exposure to the combination of temperature, pressure, humidity, radiation, and dynamic effects expected during the required operational period.

  SSCs important to safety are classified and are designed in accordance with the codes and classifications indicated in FSAR Sections 3.2 and 17A.

  FSAR Chapter 3 provides the details of the environmental activities and dynamic effects to which the SSCs important to safety are designed.

  The leak-before-break methodology demonstrates that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping. It has been applied to CPNPP Units 1 and 2 to exclude from the design basis the dynamic effects of postulated pipe ruptures in the primary coolant loop piping and 10 inch and larger reactor coolant loop branch lines, as discussed in FSAR Sections 3.6B.2.5.1 and 2.6B.2.5. Implementation of this technology eliminates the need for pipe whip restraints and jet impingement barriers. Containment design, emergency core cooling (ECC), and environmental qualification requirements are not influenced by this modification.

- GDC-10 is described in FSAR Section 3.1.2.1, Reactor Design.

  The reactor core and associated coolant, control, and protection systems are designed with adequate margins to:

  1. Assure that fuel damage is not expected during normal core operation and operational transients (Condition I) or any transient conditions arising from occurrences of moderate frequency (Condition II).

  2. Ensure return of the reactor to a safe state following infrequent incident (Condition III) events with only a small fraction of fuel rods damaged, although sufficient fuel damage might occur to preclude immediate resumption of operation.

  3. Assure that the core is intact with acceptable heat transfer geometry following transients arising from occurrences of limiting faults (Condition IV).

  FSAR Chapter 4 discusses the design bases and design evaluation of reactor components.

The RPV internals and core support structure analyses are discussed in CPNPP Units 1 and 2 FSAR Sections 3.9.N.1.4, 3.9N.2.3, 3.9N.2.4, 3.9N.2.5, 3.9N.2.6, and 3.9N.5.

FSAR Section 3.9N.5.1 describes the reactor vessel internals (RVIs) in detail. It states in part that:

- The components of the reactor internals are divided into three parts consisting of the lower core support assembly (including the entire core barrel and neutron shield pad assembly), the upper core support assembly and the in-core instrumentation support structure. The reactor internals support the core, maintain fuel alignment, limit fuel assembly movement, maintain alignment between fuel assemblies and control rod drive mechanisms (CRDMs), direct coolant flow past the fuel elements, direct coolant flow to the pressure vessel head, provide gamma and neutron shielding, and provide guides for the in-core instrumentation. Refer to FSAR Section 3.9N.5 for detailed information pertaining to the RPV internals and core structures.

- The major containment and support member of the reactor internals is the lower core support assembly, shown in FSAR Figure 3.9N-9. This assembly consists of the core barrel, the core baffle, the lower core plate and support columns, the neutron shield pads, and the core support, which is welded to the core barrel. All the major material for this assembly is Type 304 stainless steel. The lower core support assembly is supported at its upper flange from a ledge in the reactor vessel flange and its lower end is restrained in its transverse movement by a radial support system attached to the vessel wall. Within the core barrel are an axial baffle and a lower core plate, both of which are attached to the core barrel wall and form the enclosure periphery of the assembled core. The lower core support assembly and, principally, the core barrel serve to provide passageways and control for the coolant flow. The lower core plate is positioned at the bottom level of the core below the baffle plates and provides support and orientation for the fuel assemblies.

- The neutron shield pad assembly consists of four pads that are bolted and pinned to the outside of the core barrel. These pads are constructed of Type 304 stainless steel and are approximately 48 inches wide by 148 inches long by 2.8 inches thick. The pads are located azimuthally to provide the required degree of vessel protection. Additional details of the neutron shield pads are given in WCAP-7870 (1972).

- The upper core support assembly, shown in FSAR Figures 3.9N-10 and 3.9N-11, consists of the upper support, the upper core plate, the support columns, and the guide tube assemblies. The support columns establish the spacing between the upper support and the upper core plate. They are fastened to the top and bottom of these plates. The support columns transmit the mechanical loadings between the two plates and serve the supplementary function of supporting thermocouples. The guide tube assemblies sheath and guide the control rod drive shafts and control rods. They are fastened to the upper support and are restrained by pins in the upper core plate for proper orientation and support.

FSAR Section 3.9N.5.3 discusses design loadings for the RVI. It states in part that:

- The combination of design loadings fit into the normal, upset, emergency, or faulted conditions as defined in the ASME Code, Section III.
- Loads and deflections imposed on components due to shock and vibration are determined analytically and experimentally in both scaled models and operating reactors. The cyclic stresses due to these dynamic loads and deflections are combined with the stresses imposed by loads from components weights, hydraulic forces, and thermal gradients for the determination of the total stresses of the internals.
- The reactor internals are designed to withstand stresses originating from various operating conditions as summarized in FSAR Table 3.9N-1.

FSAR Section 3.9N.2.3 describes the modeling and analyses performed for dynamic response analysis of reactor internals under operational flow transients and steady-state conditions.

FSAR Section 5.2.3.1 states in part that typical material specifications used for reactor vessel internals required for ECC, for any mode of normal operation or under postulated accident conditions, and for core structural load bearing members are listed in FSAR Table 5.2-3.

2.2.3.2 Technical Evaluation

2.2.3.2.1 Introduction

The RPV internal system consists of the reactor vessel, reactor internals, fuel, and CRDMs. The reactor internals functional description is provided in the following text. The reactor internals are designed to withstand forces due to normal, upset, emergency, and faulted conditions.

Changes in the primary coolant system operating conditions (such as increase in power) also produce changes in the boundary conditions. This includes loads and temperatures experienced by the reactor internals structures or components. Ultimately, this results in changes in the stress levels in these components and changes in the relative displacement between the reactor vessel and the reactor internals. To ensure that the reactor internal components maintain their design functions, and to ensure safety questions have been reviewed, a systematic evaluation of the reactor components has been performed to assess the impact of increased core power on the reactor internal components. The reactor internal core support components are classified as follows:

Upper core support assembly (comprised of the following individual components)

- Upper support plate
- Upper core plate
- Upper core plate fuel pins
- Upper support column

Lower core support assembly (comprised of the following individual components)

- Lower support plate
- Lower core plate
- Lower core plate fuel pin
- Lower support column
- Core barrel assembly
- Baffle former assembly
- Radial keys and clevis insert assembly
- Upper core plate alignment pin

The internal structures are defined as all structures within the reactor vessel that are not core support structures, fuel assemblies, control assemblies, or instrumentation. These structures are attached to and supported by the core support structures.

Reactor Internals Functional Description

The reactor internals core support structures are within the confines of the reactor vessel. The function of the structure is to provide the direct support and restraint of the core, that is, fuel assemblies. In addition, the total structure, which includes internal structures, should provide the following:

- The orientation of the reactor core

- The orientation, guidance, and protection of the reactor control rod assemblies

- A passageway for directional and metered control of the reactor coolant flow through the reactor core

- A passageway, support, and protection for any in-vessel or in-core instrumentation

- A secondary core support for limiting the downward displacement of the core support structure in the event of a postulated failure of the core-barrel subassembly

- Reactor vessel neutron shielding

Function of Core Support Structures

<u>Upper Core Support Assembly</u>

The upper core support assembly provides the vertical and lateral restraint and lateral alignment to the top of the core through its primary components (the upper support subassembly, support columns, and the upper core plate) and its interface with the reactor vessel. The assembly also provides the support for the internal structures, such as the instrumentation conduit and supports, and reactor control rod guide tubes.

The upper support subassembly, which is supported on the outer edges, transfers the loading of the upper core support assembly to the reactor vessel. Keyways, with customized inserts to maintain required gaps, are located in the outer edges of the subassembly to provide the upper-core-support-assembly to reactor vessel to lower-core-support-assembly alignment, and to limit any transverse or rotational movement of the subassembly. There are penetrations through the subassembly for spray nozzles that allow limited flow into the reactor vessel upper head region.

The support columns transfer vertical and lateral loads to the upper support subassembly and support the upper core plate vertically. Guides are provided at the lower end of the columns for coolant flow.

The upper core plate, which is attached to the bottom of the upper support columns, forms the upper periphery of the core, transfers core loading to the support columns, and, when in place within the reactor vessel, rests on the fuel assembly springs causing the core preload. The plate is perforated to allow coolant flow while maintaining a uniform velocity profile. The underside of the plate contains the upper fuel pins, which engage the top of the fuel assemblies. The upper-core-periphery to lower-core-periphery alignment is provided through keyways in the outer edges of the plate that contain customized inserts that provide the required pin engagement gaps. In addition, the keyway/insert system limits any rotation or translation of the upper core plate.

Lower Core Support Assembly

The lower core support assembly is the major supporting assembly of the total structure. The assembly functions are as follows:

- Support the core and the attached internal structures
- Transfer these and other design loadings to the reactor vessel
- Provide the restraint and alignment of the core
- Provide the directional and metered control of the reactor coolant flow through the core
- Provide neutron shielding for the reactor vessel

Fuel assemblies are placed into the core barrel subassembly and rest on the lower core plate. The lower core plate is supported on the lower core barrel ledge and by the lower support columns, and contains the lower fuel pins that provide location and alignment for the bottom of the fuel assemblies. The lower core plate is perforated to allow directional and metered control of flow of the reactor coolant and is attached to the core barrel and the flange, forming the core barrel subassembly. The function of the core barrel subassembly is to transmit the loading to the reactor vessel. This is accomplished by the core barrel flange, which rests on a ledge provided on the reactor vessel and limited loading is transmitted at the bottom by the radial support system.

The radial support system consists of keys that are attached to the lower end of the core barrel subassembly on the lower support plate and that engage clevises provided in the reactor vessel. This system restricts the lower end of the core barrel subassembly from rotational or tangential movement, but allows for radial thermal growth and axial displacement.

Inside the core barrel, above the lower supporting component, is the baffle assembly. This subassembly forms a radial periphery of the core and, through the dimensional control of the cavity (that is, the gap between the fuel assemblies and baffle plates), provides directional and metered control of the reactor coolant through the core.

2.2.3.2.2 Input Parameters, Assumptions, and Acceptance Criteria

The principal input parameters utilized in the analysis of the reactor internal components and RPV system are the RCS design parameters provided in LR Section 1.1. For structural analysis/evaluations, the nuclear steam supply system (NSSS) design transients discussed in LR subsection 2.2.6, NSSS Design Transients were considered. The fuel considered is a full core of Westinghouse 17x17 VANTAGE+ fuel assembly with intermediate flow mixer (IFM) grids with thimble plugging devices installed.

The acceptance criteria are:

- The design core bypass flow limit with the thimble plugging devices installed is 5.8 percent of the total vessel flow rate.

- The rod cluster control assembly (RCCA) drop time Technical Specification of 2.7 seconds.

- For the structural and fatigue evaluations of core support components, the components stresses meet the allowable stress limits and the cumulative fatigue usage factors must be less than 1.0.

- Reactor Internals Heat Generation

  The presence of radiation-induced heat generation rates in the reactor internals components, in conjunction with the reactor coolant fluid temperatures, results in thermal gradients within and between the components. The resultant material temperature gradients cause thermal stresses and thermal growth that must be considered in the design and analysis of the various components. The primary design considerations are to ensure that thermal growth is consistent with the functional requirements of the components and to ensure that the applicable American Society of Mechanical Engineers (ASME) Code requirements are satisfied as part of the components evaluation.

  The reactor internals components subjected to significant radiation-induced heat generation are the core baffle plates, former plates, core barrel, baffle-former bolts, barrel-former bolts, neutron pads, and the upper and lower core plates. However, due to

the relatively lower heat generation rates in the neutron pad materials, this component experiences little, if any, temperature rise relative to the surrounding reactor coolant.

This section provides a description of the methodology that is used to determine the radiation-induced heat generation rates in the internals components and provides the results of the evaluation assessing the impact of the CPNPP Units 1 and 2 core power uprate to 3,612 MWt on these heating rates. The current evaluation included the impact on heat generation rates in the core baffle, barrel, neutron pad, and the upper and lower core plates. The impact on the heat generation rates applicable to the core formers is inferred from the data generated for the core baffle and barrel.

2.2.3.2.2.1 Description of the Evaluation of Heat Generation Rates

The initial design of the CPNPP Units 1 and 2 reactor internals was based on a reactor core operating at 3,565 MWt with radial and axial core power distributions designed to produce conservative results for components located radial outboard of the reactor core (baffle plates, barrel, and neutron pad) as well as for the core components located above and below the core (upper and lower core plates). These baseline heat generation rates were documented in WCAP-9620, Revision 1 (Reference 1).

Over time it was noted that, in general, the heat generation rates calculated for the radial components remained conservative. One of the primary reasons for this observation is that the tendency within the industry to transition from out/in fuel management to low leakage core designs has resulted in greatly reduced heat generation at the core periphery which, in turn, has provided significant margin relative to the calculated design heating rates. For the core plates, however, evolving fuel assembly designs and core operation resulted in a tendency toward increased heating rates in these axial components. As a result, a new set of conservative heat generation rates were calculated for use in the assessment of upper and lower core plate structural integrity. These updated values were intended to supersede the core plate data provided in Reference 1 and to establish a new baseline for upper and lower core plate structural analysis.

For the evaluation of the CPNPP Units 1 and 2 radial internals components at the uprated core power of 3,612 MWt, it was anticipated that, due to the implementation of low leakage fuel cycle designs, the heating rates used in the original analysis would remain bounding at the uprated core power. To test this hypothesis, a set of heat generation rates based on the projected uprate fuel management strategy was calculated and the results compared directly with the corresponding data provided in Reference 1. The results of this comparison are discussed below. The heat generation rate calculation for the radial components was completed using the DORT two-dimensional discrete ordinates code from the DOORS 3.2 Code Package (Reference 2). This combination of codes has been used to support numerous pressure vessel fluence evaluations and is accepted by the Nuclear Regulatory Commission (NRC) for deterministic particle transport calculations. The transport cross-sections used in the calculations were taken from the BUGLE-96 coupled neutron/photon cross-section library (Reference 3) that was generated specifically for light water reactor (LWR) applications.

In the case of the CPNPP Units 1 and 2 radial internals, two sets of heating rate calculations were completed in order to accurately model the components being evaluated. The core baffle plates were analyzed using an x, y geometric model and the core barrel and neutron pad heating rates were determined using an r,θ geometry. All of the transport calculations used a P5 expansion of the neutron scattering cross-sections and an S16 order of angular quadrature.

For the evaluation of the upper and lower core plates, no additional calculations were required. The data supplied as the updated baseline for axial component heating represent a clearly bounding situation relative to both core power level and axial core power distribution. In the analysis for the 4-loop internals designs, the assumed core power level was 3,950 MWt and the axial distribution of core power was assumed to be uniform throughout the reactor core. Thus, the core power assumption in the baseline data exceeds that of the CPNPP Units 1 and 2 uprate conditions by several hundred megawatts and, in addition, the assumption of a uniform axial power distribution increases the leakage from the top and bottom of the reactor core relative to any realistic core operating conditions.

2.2.3.2.2.2 Results of the Evaluation of Heat Generation Rates

The results of the radiation-induced heat generation rate evaluations are provided in Tables 2.2.3-1 through 2.2.3-3. In Table 2.2.3-1, a comparison of the radial component heating rates applicable to the CPNPP Units 1 and 2 reactors operating at 3,612 MWt with the corresponding data from Reference 1 is provided. This comparison shows that, for all of the radial components, significant margin exists between the current calculations and the original design values. The applicable heating rates for the upper and lower core plates are provided in Tables 2.2.3-2 and 2.2.3-3, respectively. These data represent bounding heat generation rates for the upper and lower core plates.

2.2.3.2.2.3 Conclusions

The component radiation-induced heating rates applicable to CPNPP Units 1 and 2 reactors operating at a core power level of 3,612 MWt are summarized in Tables 2.2.3-1 through 2.2.3-3. The results provided in these tables show significant margin between current values and the original heating rate calculations for the radial components due to the implementation of low leakage fuel management. The previously determined baseline heat generation rates for the upper and lower core plates have also been shown to remain bounding.

2.2.3.2.3 Description of Analyses and Evaluations

The RVIs have been analyzed for the CPNPP Units 1 and 2 SPU revised design parameters and the design basis load combinations. The analysis of the components was performed for the normal, upset, emergency, and faulted conditions (LOCA/seismic). The results of these analyses confirm that there is no adverse impact on the structural adequacy of the reactor internals components for the SPU conditions.

Thermal-Hydraulic System Evaluations

System Pressure Losses

The principal RCS flow route through the RPV system at CPNPP Units 1 and 2 begins at the inlet nozzles. At this point, flow turns downward through the reactor vessel/core barrel annulus. After passing through this downcomer region, the flow enters the lower reactor vessel dome region. This region is occupied by the internals energy absorber structure, lower support columns, bottom-mounted instrumentation columns, and supporting tie plates. From this region, flow passes upward through the lower core support plate and into the core region. After passing upward through the core, the coolant flows into the upper plenum, turns, and exits the reactor vessel through the outlet nozzles. Note that the upper plenum region is occupied by support columns and RCCA guide columns.

A key area in the evaluation of core performance is the determination of hydraulic behavior of coolant flow within the reactor internals system, that is, vessel pressure drops, core bypass flows, RPV fluid temperatures, and hydraulic lift forces. The pressure loss data are necessary input to the LOCA and non-LOCA safety analyses and to overall NSSS performance calculations. The hydraulic forces are considered in the assessment of the structural integrity of the reactor internals, core clamping loads generated by the internals hold-down spring, and the stresses in the reactor vessel closure studs.

Thermal-hydraulic evaluations were performed by solving the mass and energy balances for the reactor internals fluid system. These analyses determined the distribution of pressure and flow within the reactor vessel, reactor internals, and the reactor core. Results were obtained with a full core of Westinghouse 17x17 VANTAGE+ fuel with IFM grids with thimble plugging devices installed, and at RCS conditions, as given in Licensing Report (LR) Section 1.1.

Bypass Flow Analysis

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. Variations in the size of some of the bypass flow paths, such as gaps at the outlet nozzles and the core cavity, occur during manufacturing or change due to fuel assembly changes. Plant-specific, as-built dimensions are used in order to demonstrate that the core bypass flow limits are not violated. Therefore, analyses are performed to estimate core bypass flow values to either show that the design bypass flow limit for the plant are not exceeded or to determine a revised design core bypass flow.

Fuel assembly hydraulic characteristics and system parameters, such as inlet temperature, reactor coolant pressure, and flow were used to determine the impact of SPU RCS conditions on the total core bypass flow. The results of this analysis calculated a core bypass flow value of 4.94 percent with the thimble plugging devices installed. Therefore, the design core bypass flow value of 5.8 percent with thimble plugging devices installed remains acceptable.

Hydraulic Lift Forces

An evaluation was performed to estimate hydraulic lift forces on the various reactor internal components for the SPU parameters shown in LR Section 1.1. This is done to show that the reactor internals assembly would remain seated and stable for all conditions. Based on the evaluation performed for the CPNPP Units 1 and 2 SPU, the reactor lower internals remain seated and stable for the following SPU, RCS conditions:

- Hot full-power normal conditions
- Cold zero-power normal conditions
- Seismic operating basis earthquake (OBE) with hot full-flow upset conditions
- Hot pump overspeed (HPO) upset conditions (without OBE)

In addition, a minimum of 100,000-pound hold-down force is maintained during normal operating conditions. These evaluations conservatively assume that no internals hold-down contribution is provided by the fuel assemblies. For HPO with OBE upset conditions, the lower internals liftoff the vessel ledge assuming that all the fuel assemblies liftoff. The liftoff of the lower internals due to HPO with OBE is not considered to be a safety concern. In order to confirm this, evaluations were performed for the increase in stress in the hold-down spring for the HPO+OBE condition and was shown to be acceptable because the corresponding tensile stress in the hold-down spring is approximately 47 percent of the yield stress (Sy). Therefore the increased stress in the hold-down spring due to the HPO + OBE condition is still within the elastic range (less than yield stress) and the momentary liftoff condition will not lead to plastic deformation of the hold-down spring. The hold-down spring will remain elastic and be able to return to the initial pre-load condition.

Upper Head Fluid Temperatures

The average temperature of the primary coolant fluid that occupies the reactor vessel closure head (RVCH) volume is an important initial condition for certain dynamic LOCA analyses. Therefore, it was necessary to determine the upper head temperature when changes in the RCS conditions take place in the plant. Determination of upper head temperature stemmed from the Thermal Hydraulic Reactor Internals Vessel Evaluation (THRIVE) code calculations used to assess the core bypass flow. The THRIVE code models the interaction between all different flow paths into and out of the closure head region. Based on this interaction, it calculates the core bypass flow into the head region and the average head fluid temperature for different flow path conditions. CPNPP Units 1 and 2 are designed such that the upper head region is at T_{cold} and at SPU conditions the calculated upper head region average fluid temperature remained at T_{cold}. These upper head fluid temperatures were provided as inputs and were used in subsequent LOCA analyses.

RCCA Scram Performance Evaluation

The RCCAs represent a critical interface between the fuel assemblies and the other internal components. It is imperative to show that the SPU RCS conditions do not adversely impact the operation of the RCCAs, either during accident conditions or normal operation.

The analysis performed determined the potential impact of the conditions shown in LR Section 1.1 on the limiting RCCA drop time. The maximum estimated RCCA drop time was calculated to be 2.47 seconds to the top of dashpot. Therefore, the limit of 2.7 seconds is acceptable.

Mechanical System Evaluations

LOCA Loads

To perform the RPV LOCA analyses of CPNPP Units 1 and 2, a finite element model of the RPV system was developed. The mathematical model of the RPV is a three-dimensional, nonlinear finite element model that represents the dynamic characteristics of the reactor vessel and its internals in the six geometric degrees of freedom. For the CPNPP Units 1 and 2 SPU, LOCA analyses were performed to generate core plate motions and the reactor vessel/internals interface loads.

The results of LOCA reactor vessel displacements and the impact forces calculated at vessel/internals interfaces are used to evaluate the structural integrity of the reactor vessel and its internals. The core plate motions were used in the fuel grid crush analysis and to confirm the structural integrity of the fuel as discussed in detail in this LR subsection 2.8.1, Fuel System Design.

Seismic Analyses

The SPU does not impact the seismic response of the reactor internals. Therefore, the nonlinear time-history seismic analysis of the RPV system was not performed.

Flow-Induced Vibrations

Flow-induced vibrations of pressurized water reactor internals have been studied in the industry for a number of years. The objective of these studies is to show the structural integrity and reliability of reactor internal components. These efforts have included in-plant tests, scale-model tests, as well as tests in fabricators' shops and bench tests of components, and various analytical investigations. The results of these scale-model and in-plant tests indicate that the vibrational behavior of two-, three-, and four-loop plants is essentially similar, and the results obtained from each of the tests compliment one another and allow a better understanding of the FIV phenomena. Based on the analysis performed for CPNPP Units 1 and 2, reactor internals response due to FIV is extremely small and well within the allowable based on the high cycle endurance limit for the material. The results of FIV analyses for the CPNPP Units 1 and 2 SPU are provided in Tables 2.2.3-4 and -5.

Evaluation of Reactor Internal and Core Support Structure Components

In addition to supporting the core, a secondary function of the RVI assembly is to direct coolant flows within the vessel. While directing primary flow through the core, the internals assembly also establishes secondary flow paths for cooling the upper regions of the reactor vessel and

the internals structural components. Some of the parameters influencing the mechanical design of the internals lower assembly are the pressure and temperature differentials across its component parts and the flow rate required to remove heat generated within the structural components due to radiation (for example, gamma heating). The configuration of the internals provides adequate cooling capability. Also, the thermal gradients resulting from gamma heating and core coolant temperature changes are maintained below acceptable limits within and between the various structural components.

The Units 1 and 2 reactor internals were designed and built prior to the implementation of Subsection NG of the ASME Boiler and Pressure Vessel Code. Therefore, a plant-specific stress report on the reactor internals was not required. The structural integrity of the Units 1 and 2 reactor internals design has been ensured by analyses performed on both generic and plant-specific bases to meet the intent of the ASME Code. These analyses were used as the basis for evaluating critical Units 1 and 2 reactor internal components for SPU RCS conditions and revised NSSS design transients.

Structural evaluations demonstrate that the structural integrity of reactor internal components is not adversely affected either directly by the SPU RCS conditions and NSSS design transients, or by secondary effects on reactor thermal-hydraulic or structural performance. Heat generated in reactor internal components, along with the various fluid temperature changes, results in thermal gradients within and between components. These thermal gradients result in thermal stresses and thermal growth, which must be considered in the design and analysis of the various components.

Component Evaluations

A series of evaluations for Units 1 and 2 were performed on reactor internal components for the SPU conditions. The most limiting reactor internal components that were evaluated are as follows:

- Upper core plate
- Lower support plate
- Lower core plate
- Lower support column
- Core barrel
- Baffle-former bolts

The results of these evaluations demonstrate that the above listed components are structurally adequate for the SPU conditions and the fatigue usage factors were less than 1.0. Since the surface stress range factor does not significantly increase for each component and the most limiting components qualify, the remaining core support components qualify as well. A summary of stresses versus allowable and corresponding fatigue usage factors is given in Table 2.2.3-6.

Results

Analyses have been performed to assess the effect of changes due to the SPU at CPNPP Units 1 and 2. The various results reached are as follows:

- The design core bypass flow value of 5.8 percent of the total vessel flow with thimble plugging devices installed is maintained for the SPU conditions.
- An RCCA performance evaluation was completed and the results indicated that the RCCA drop time limit of 2.7 seconds is acceptable for the SPU effort.
- Evaluations of the limiting reactor internal core support components were performed, which indicated that the structural integrity of the reactor internals is maintained at the SPU conditions and the cumulative fatigue usage factors were all shown to be less than 1.0.

The results of component structural analyses are summarized in Table 2.2.3-6.

2.2.3.3 Conclusion

Luminant Power has reviewed the evaluations related to the structural integrity of reactor internals and core supports and concludes that the evaluations have adequately addressed the effects of the proposed SPU on the reactor internals and core supports. Luminant Power further concludes that the evaluations have demonstrated that the reactor internals and core supports will continue to meet the requirements of 10 CFR 50.55a, GDC-1, -2, -4, and -10 following implementation of the proposed SPU. Therefore, Luminant Power finds the proposed SPU acceptable with respect to the design of the reactor internal and core supports structures.

2.2.3.4 References

1. WCAP-9620, Revision 1, "Reactor Internals Heat Generation and Neutron Fluences," December 1983.

2. RSICC Computer Code Collection CCC-650, "DOORS 3.2, One-, Two-, and Three-Dimensional Discrete Ordinates Neutron/Photon Transport Code System," April 1998.

3. RSIC Data Library Collection DLC-185, "BUGLE-96, Coupled 47 Neutron, 20 Gamma Ray Group Cross-Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications," March 1996.

Table 2.2.3-1 Comparison of Original Zone Average Heating Rates with Current Analysis		
	Region Average Long-Term Heating Rates (Btu/hr-lbm)	
Location	WCAP-9620 Analysis (Ref. 1)	Current Analysis
Baffle Plate 18	945	724
Baffle Plate 19	1,070	627
Baffle Plate 20	996	516
Baffle Plate 21	802	359
Core Barrel	186	111
Neutron Pad	37.8	16.5

Table 2.2.3-2

Spatial Distribution of Long-Term Heating Rates in the 3.00-inch Upper Core Plate – Reactor Power = 3,950 MWt

Heat Generation Rate (BTU/hr-lbm)

Radial Mesh Midpoint (inches)	Bottom Surface 0.00	0.25	Distance Through Plate (inches) 0.75	1.25	1.75	2.25	2.75	Top Surface 3.00
0.98	479	432	339	270	218	178	148	133
2.95	479	432	338	269	217	178	148	133
4.92	478	431	337	268	216	177	147	132
6.89	477	430	335	267	215	176	147	132
8.86	475	428	334	266	215	176	146	132
10.83	473	427	333	265	214	175	146	131
12.80	472	425	332	264	213	175	145	131
14.76	471	424	331	263	213	174	145	130
16.73	469	423	330	263	212	174	145	130
18.70	468	422	330	262	212	173	144	130
20.67	467	421	329	262	211	173	144	130
22.64	467	421	329	262	211	173	144	130
24.61	467	421	329	262	211	173	144	130
26.57	467	421	329	262	211	173	144	130
28.54	467	421	329	262	211	173	144	130
30.51	467	421	329	262	211	173	144	130
32.48	467	421	329	262	211	173	144	129
34.45	466	420	328	262	211	173	144	129
36.42	464	418	327	261	210	172	143	129
38.39	461	416	325	259	209	171	142	128
40.35	457	412	322	257	207	169	141	127
42.32	451	406	318	253	204	167	139	125
44.29	442	399	312	248	200	164	136	122
46.26	431	388	304	242	195	160	133	119
48.23	416	375	293	234	188	154	128	115
50.20	396	358	280	223	179	147	122	109
52.17	372	336	263	209	169	138	114	103
54.13	344	310	243	193	156	127	106	95
56.10	312	281	220	175	141	116	96	86
58.07	278	251	196	156	126	103	86	77
60.04	243	219	171	137	110	90	75	67
62.01	207	187	146	116	94	77	64	57
63.78	175	157	123	98	79	65	54	48
64.96	152	137	107	85	68	56	46	41
65.65	136	123	96	76	61	50	41	37
66.15	123	111	86	68	55	45	37	33
66.64	91	83	67	53	43	35	29	26
67.20	64	59	48	39	31	26	22	20
67.89	55	49	36	29	23	19	16	15
68.70	54	46	31	23	18	14	13	12
69.52	53	45	29	20	14	12	10	10
70.33	51	43	27	18	13	10	9	8
71.15	48	41	25	17	12	9	7	7
71.96	44	37	23	15	10	8	7	6
72.78	40	34	21	14	9	7	6	5
73.59	35	29	18	12	8	6	5	5
74.00	33	27	17	11	7	5	5	4

Table 2.2.3-3

Spatial Distribution of Long-Term Heating Rates in the 2.00" Lower Core Plate – Reactor Power = 3,950 MWt

Heat Generation Rate (BTU/hr-lbm)

Radial Mesh Midpoint (inches)	Bottom Surface 0.00	Distance Through Plate (inches) 0.25	0.75	1.25	1.75	Top Surface 2.00
0.98	694	782	958	1196	1518	1679
2.95	693	780	956	1196	1522	1684
4.92	693	781	956	1197	1524	1687
6.89	690	778	953	1193	1519	1683
8.86	686	773	946	1185	1507	1668
10.83	680	766	939	1174	1493	1652
12.80	676	761	932	1165	1482	1641
14.76	672	757	927	1159	1474	1631
16.73	670	755	924	1156	1470	1628
18.70	669	753	922	1153	1467	1624
20.67	667	751	919	1150	1463	1619
22.64	665	749	916	1146	1458	1613
24.61	665	748	915	1144	1455	1611
26.57	667	750	918	1148	1460	1616
28.54	670	755	924	1157	1471	1628
30.51	675	760	932	1166	1484	1642
32.48	677	764	936	1173	1492	1651
34.45	678	765	937	1174	1493	1653
36.42	678	764	936	1172	1491	1650
38.39	677	763	935	1171	1490	1649
40.35	678	764	937	1172	1492	1652
42.32	679	766	941	1178	1500	1660
44.29	681	769	945	1185	1508	1670
46.26	679	768	946	1187	1511	1674
48.23	670	759	937	1177	1499	1660
50.20	650	737	912	1146	1460	1617
52.17	616	700	866	1090	1388	1537
54.13	567	644	798	1004	1279	1417
56.10	505	573	708	890	1134	1256
58.07	434	491	604	758	965	1068
60.04	359	405	496	621	788	872
62.01	286	321	391	488	618	683
63.78	224	251	304	377	476	525
64.96	186	207	249	308	386	425
65.65	163	180	216	266	331	363
66.15	144	160	191	236	292	320
66.64	120	133	158	197	253	280
67.20	98	107	127	161	216	244
67.89	79	87	103	134	185	211
68.70	64	71	84	111	157	180
69.52	54	59	70	93	135	156
70.33	45	49	58	79	117	136
71.15	38	42	49	67	102	120
71.96	32	35	42	58	89	105
72.78	26	29	35	49	76	90
73.59	22	24	28	40	64	76
74.00	19	21	25	36	58	69

Table 2.2.3-4 Lower Internal Critical Component Stresses Due to FIV		
Component	**Maximum Alternating Stress psi**	**ASME Code Endurance Limit(1) (high-cycle fatigue) psi**
Core Barrel Flange	[ ]a,c	23,700
Core Barrel Girth Weld	[ ]a,c	23,700

Note:

1. Basis is ASME Code section NB-3222 and Figure I-9.2.2, Curve A and Table I-9.2.2.

Table 2.2.3-5 Upper Internal Critical Component Strains Due to FIV		
Component	**Uprate Mean Strain in/in x 10^{-6}**	**Endurance Limit Strain in/in x 10^{-6}**
Guide Tubes	[ ]a,c	101.5

Table 2.2.3-6

Reactor Internal Components Stresses and Fatigue Usage Factors

Component	Stress Intensity (ksi) S.I. = (P_m + P_b + Q)	Allowable S.I. (3 S_m) ksi	Fatigue Usage
Upper Core Plate	[ ]a,c	48.6	[ ]a,c
Lower Support Plate	[ ]a,c	48.3	[ ]a,c
Lower Core Plate	[ ]a,c	48.6	[ ]a,c
Lower Support Columns	[ ]a,c	48.3	[ ]a,c
Core Barrel Outlet Nozzle:			
Section A-A	[ ]a,c	34.4	[ ]a,c
Section B-B	[ ]a,c	49.2	[ ]a,c
Baffle-Former Bolts(2)	--	--	--

Notes:

1. Exceeded 3 S_m limit, simplified elastic-plastic analysis performed to calculate fatigue usage.
2. The basis of the baffle-former bolt qualification is a fatigue test. The evaluation of the revised loads consisted of demonstrating that the loads associated with SPU are acceptable for the plant design life. Therefore, it is concluded that the baffle-former bolts are structurally adequate for the SPU RCS conditions.

2.2.4 Safety-Related Valves and Pumps

2.2.4.1 Regulatory Evaluation

Safety-related pumps and valves typically designated as Class 1, 2, or 3 under Section III of the American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel (B&PV) Code and within the scope of Section XI of the ASME Boiler & Pressure Vessel Code and the ASME Operations and Maintenance Code, as applicable, were reviewed. The review focused on the effects of the proposed stretch power uprate (SPU) on the required functional performance of the valves and pumps. The review also covered any impacts that the proposed SPU may have on the licensee's motor-operated valve programs related to Generic Letter (GL) 89-10, GL 96-05, and GL 95-07. Lessons learned from the motor-operated valve program and the application of those lessons learned to other safety-related power-operated valves was evaluated. The Nuclear Regulatory Commission's (NRC's) acceptance criteria are based on:

- General Design Criterion (GDC)-1, insofar as it requires that structures, systems, and components (SSCs) important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.
- GDCs-37, -40, -43, and -46, insofar as they require that the emergency core cooling system (ECCS), the containment heat removal system, the containment atmospheric cleanup systems, and the cooling water system, respectively, be designed to permit appropriate periodic testing to ensure the leak-tight integrity and performance of their active components.
- GDC-54, insofar as it requires that piping systems penetrating containment be designed with the capability to periodically test the operability of the isolation valves to determine if valve leakage is within acceptable limits.
- 10 CFR 50.55a(f), insofar as it requires that pumps and valves subject to that section must meet the in-service testing program requirements identified in that section.

Current Licensing Basis

The adequacy of the Comanche Peak Nuclear Power Plant (CPNPP) design relative to the GDC is discussed in Final Safety Analysis Report (FSAR) Section 3.1. Specifically, the adequacy of CPNPP design relative to conformance to:

- GDC-1, Quality Standards and Records, is described in FSAR Section 3.1.1.1.

  SSCs important to safety are designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Quality standards applicable to safety related SSCs are generally contained in codes such as the ASME B&PV Code. The applicability of these codes is specifically identified throughout the FSAR and is summarized in FSAR Section 3.2.5.

FSAR Chapter 17 provides direct reference to the Quality Assurance Program established to provide assurance that safety related SSCs satisfactorily perform their intended safety functions. The procedures for generating and maintaining appropriate design, fabrication, erection, and testing records are contained within the referenced documents.

- GDC-37, Testing of Emergency Core Cooling System, is described in FSAR Section 3.1.4.8.

  Each active component of the ECCS can be individually actuated on the normal power source and can be transferred to the emergency power source at any time during appropriate plant periodic tests. Tests can also be performed during shutdown to demonstrate proper automatic operation of the ECCS.

- GDC-40, Testing of Containment Heat Removal System, is described in FSAR Section 3.1.4.11.

  System piping, valves, pumps, heat exchangers, and other components of the containment heat removal system are arranged so that each component can be tested periodically for operability, including transfer to emergency power sources. The delivery capability of the containment spray system is tested periodically, to the extent practical, up to the last isolation valve before the spray nozzles. The delivery capability of the spray nozzles is tested periodically by blowing low pressure air through the nozzles and verifying the flow. The containment spray systems are tested for operational sequence as close to the design as practical.

- GDC-43, Testing of Atmosphere Cleanup Systems is described in FSAR Section 3.1.4.14.

  The containment atmosphere cleanup system can be tested as follows:

  - The operation of the spray pumps can be tested by recirculation through a test line to the refueling water storage tank (RWST). The system valves can be operated through their full travel, and the system can be checked for leaktightness during testing.

  - The hydrogen purge system can be tested periodically to demonstrate its ability to function.

- GDC-46, Testing of Cooling Water System, is described in FSAR Section 3.1.4.17.

  The design provides for testing of operating system components, which are interchanged periodically, for operability and functional performance.

  Redundancy and isolation are provided to allow periodic pressure and functional testing of the system as a whole, including the functional sequence that initiates system

operation. At any time during reactor operation, each component of the system can be mechanically connected to the preferred power source to demonstrate operability.

- GDC-54, Piping Systems Penetrating Containment, is described in FSAR Section 3.1.5.5.

  Piping systems penetrating the primary reactor containment are provided with containment isolation valves. Fittings are provided to permit periodic leakage-rate testing of isolation valves to ensure that leakage is within the acceptable limit. Penetrations that must be closed for containment isolation have redundant valves and associated apparatus. Each valve is tested periodically during normal operation or during shutdown conditions to ensure its operability when needed.

Inservice Testing of Safety-Related Pumps and Valves

As addressed in FSAR Sections 3.9N.6 and 3.9B.6, the in-service testing of pumps and valves is documented in the CPNPP Inservice Testing Plan.

Inservice Testing Plan for Pumps and Valves, hereafter referred to as the in-service testing (IST) plan, has been prepared to summarize the test program for certain pumps and valves pursuant to the requirements of the Code of Federal Regulations, 10 CFR 50.55a(f)(4); and as modified by Relief Request A-1, "Request for Alternative from 10 CFR 50.55a(f)(4)(i) and (ii) for Inservice Testing Frequency Under 10 CFR 50.55a(a)(3)(i)," and by the NRC Safety Evaluation Report (SER) on the CPNPP RI-IST Program. This testing plan is applicable to CPNPP Units 1 and 2.

CPNPP Technical Specification 5.5.8, Inservice Testing Program, states that the Inservice Testing Program provides controls for in-service testing of ASME Code Class 1, 2, and 3 components and that the program shall include testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda.

Containment Leakage Rate Testing

GDC-52, Capability For Containment Leakage Rate Testing, is described in FSAR Section 3.1.5.3.

The containment is designed and constructed, and the necessary equipment is provided, to permit periodic integrated leakage-rate tests during plant lifetime, in accordance with reduced pressure-test program requirements of 10 CFR 50, Appendix J.

2.2.4.2 Technical Evaluation

2.2.4.2.1 Introduction

Performance/Inservice Testing of Safety-Related Pumps and Valves

Implementation of the 1989 Edition IST Plan was completed on CPNPP Unit 1 before that unit was returned to power following the third refueling outage. This superseded the original Unit 1 Inservice Testing Plan for Pumps and Valves developed for the first inspection interval. The original Unit 1 IST Plan was implemented per the requirements of the 1986 Edition of Section XI. This 1989 Edition IST Plan constituted an update of the original Unit 1 IST Plan to a later approved Code edition as allowed by 10 CFR 50.55a(f)(4)(iv) and as approved by the NRC staff. This IST Plan was to remain in effect for Unit 1 for the 120-month interval following the date of the Unit 1 commercial operation (August 13, 1990). An exemption from regulation 10 CFR 50.55a (f)(4)(ii) to the 10-year test interval for Unit 1 was granted by the NRC on June 21, 1995. The extension allowed Unit 1 to remain under the 1989 Edition until the conclusion of the 10-year test interval for Unit 2 (August 3, 2003).

This IST Plan was in effect for Unit 2 for the 120-month interval following the date of the Unit 2 commercial operation (August 3, 1993 to August 2, 2003).

The IST Plan for Units 1 and 2 first interval end date was extended from August 2, 2003 to not later than August 2, 2004 (see TXX-03075, dated April 11, 2003).

The start date of the IST Plan second interval for Unit 1 and 2 was August 3, 2004, and the end date for the second interval is August 2, 2013 (see TXX-04134).

Within the IST Plan, a section is provided for Inservice Pump Testing Plan and a section is provided for Inservice Valve Testing Plan. These two sections of the overall IST Plan are briefly described below.

The scope of the In-service Pump Testing Plan is derived from the requirements of ASME Operations and Maintenance (OM) Code 1998 Edition through 2000 Addenda, as modified by 10CFR50.55a(f)(4) and Relief Request A-1.

The IST Plan includes pumps in the following systems:

- Auxiliary feedwater system
- Component cooling water system
- Chilled water (safety)
- Chemical and volume control system (centrifugal charging and boric acid transfer)
- Containment spray system
- Reactor makeup water system
- Diesel Generator Fuel oil transfer
- Residual heat removal system
- Spent fuel pool cooling system

- Safety injection system
- Service water system
- Safeguards building floor drain pumps

The scope of the Inservice Valve Testing Plan is derived from the requirements of ASME OM Code Subsection ISTC, Appendix I and Appendix II as modified by 10 CFR 50.55a(f)(4) and Relief Request A-1.

The IST Plan includes valves in the following systems or valve groups:

- Auxiliary feedwater
- Component cooling water
- Chilled water (safety and non-safety)
- Chemical and volume control
- Containment spray
- Demineralized and reactor makeup water
- Diesel generator auxiliaries
- Feedwater
- Main steam
- Reactor coolant
- Residual heat removal
- Spent fuel pool cooling
- Safety injection
- Service water
- Ventilation (Control Room air conditioning)
- Vents and drains
- Miscellaneous containment isolation valves
- Safety and relief valves
- Motor-operated valves (MOVs)

Air-Operated Valves (AOVs)

CPNPP has incorporated the industry guidance for AOVs that includes guidelines for design bases reviews, testing, and maintenance of AOVs. Those AOVs that are categorized as high safety significant or low safety significant components are also included in the Inservice Valve Testing Plan.

Containment Leakage Rate Testing Program

The CPNPP Containment Leakage Rate Testing Program implements testing requirements in accordance with 10 CFR 50 Appendix J, Option B, as modified by any approved exemptions, and the guidelines contained in Regulatory Guide 1.163, Performance-Based Containment Leak Test Program (dated September 1995). Regulatory Guide (RG) 1.163 endorses use of Nuclear Energy Institute (NEI) 94-01 Rev. 0 (dated July 26, 1995), Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J, with some exceptions. NEI 94-01 references use of American National Standards Institute/American Nuclear Society

(ANSI/ANS)-56.8-1994, Containment System Leakage Testing Requirements, for technical methods and techniques for Type A, B, and C tests with some exceptions.

2.2.4.2.2 Description of Analyses and Evaluations

The following addresses the impact of the proposed SPU on the performance requirements of CPNPP safety-related pumps and valves in IST Plan. The discussion is organized by system or groups of systems and the respective IST Plan for pumps and valves are discussed therein. It is noted that the IST Plan includes motor operated valves identified per GL 89-10 and AOVs that meet the industry guidelines for AOVs.

Nuclear Steam Supply System Systems

The nuclear steam supply system (NSSS) systems include the reactor coolant system, chemical and volume control system, safety injection system, residual heat removal system, and containment spray system. Evaluations show that the SPU has no or negligible impact on system operating pressures, flow rates, and pump head performance for NSSS systems under normal operating conditions.

The ECCS includes chemical and volume control system, residual heat removal system, safety injection system, and containment spray system. No changes are being made to these systems; system pressures, temperatures, and flow rates are not impacted by the SPU.

Based on these evaluations, the SPU has no impact on the performance characteristics and IST Plan requirements for safety-related pumps and valves (AOVs, MOVs, and check valves) in the NSSS systems or ECCS.

Balance-of-Plant Systems

Auxiliary Feedwater System

The feedline break analysis requires a minimum flow of 430 gpm through the TDAFP to the three intact steam generators prior to isolation of the broken feedwater line. This is an increase from the current requirement of 400 gpm. There is adequate pump margin and head capacity to accommodate this flow rate. The motor-driven auxiliary feedwater pump and associated IST Plan valves (AOVs, MOVs, and check valves) are not impacted by the SPU.

Component Cooling Water System

There are heat load changes in the component cooling water systems as a result of the SPU, but system pressures and flows are not changing. Thus, the IST Plan valves (AOVs, MOVs, and check valves) and pumps are not impacted by the SPU.

Containment Spray System

There are heat load changes in the containment spray system as a result of the SPU, but system pressures and flows are not changing. Thus, the IST Plan valves (MOVs and check valves) and pumps are not impacted by the SPU.

Chilled Water System (Safety and Non-safety)

There are negligible heat load changes in the chilled water as a result of the SPU and no system pressure or flow changes. Thus, the IST Plan valves (MOVs, check valves), and pumps are not impacted by the SPU.

Demineralized and Reactor Makeup Water System

There are no changes to the demineralized and reactor makeup water system as a result of the SPU. Thus, the IST Plan valves (AOVs and check valves) and pumps in this system are not impacted by the SPU.

Diesel Generator Auxiliaries

There are no changes to the diesel generator auxiliaries as a result of the SPU. Thus, the IST Plan valves (solenoid valves, check valves), and pumps in this system are not impacted by the SPU.

Feedwater System

Feedwater system pressure and flow increase to support the SPU. To achieve this, the steam generator feedwater pump turbine speed increases. Consequently, the shutoff head of the pump increases about 13 psi, which affects the AOV differential pressure analysis that is used to determine AOV minimum thrust/torque requirements for feedwater control valves (Unit 1 and Unit 2) and feedwater preheater bypass valves (Unit 2 only) and feedwater split flow bypass valves (Unit 2 only).

The feedwater system changes are within the design of the feedwater check valves, and the feedwater check valve performance will be acceptable for SPU. The feedwater isolation valve performance will be acceptable for SPU.

There are no IST Plan pumps in this system. The respective differential pressure analyses for the IST Plan valves (AOVs and MOVs) in this system will be updated for this increase in feedwater system pressure; but the proposed SPU does not impact IST Plan requirements for the feedwater system.

Main Steam System

With the SPU conditions, the main steam system pressure does not change for Unit 1 and decreases for Unit 2. For both units, main steam flow increases. An analysis has determined that cooldown capabilities with the main steam system safeties and atmospheric relief valves are adequate and no changes are being made to those valves. The main steam isolation valves are not adversely impacted since the closing time is based on worst case break flow conditions. These conditions are not affected by the SPU. Refer to LR subsection 2.5.5.1 main steam. There are no IST Plan pumps in this system. There is no impact to the IST Plan valves in the

main steam system. Thus, the proposed SPU does not impact IST Plan requirements for the main steam system.

Spent Fuel Pool Cooling System

The proposed SPU results in a heat load increase for the spent fuel pool. However, no changes are being made to the system flow requirements. Thus, the IST Plan valves (check valves) and pumps are not impacted by the SPU.

Service Water System

There are heat load changes in the service water system as a result of the SPU, but system pressures and flows are not changing. Thus, the IST Plan valves (MOVs and check valves) and pumps are not impacted by the SPU.

Control Room Ventilation

This system contains no IST Plan pumps and only four IST Plan check valves. There are no IST Plan power-operated valves (MOVs, AOVs, or solenoid-operated valves) in this system. These check valves are part of the instrument air system, but are included in this system because their safety function is more closely associated with the Control Room ventilation system. No changes are being made to the instrument air system check valves as part of the proposed SPU. Thus, IST Plan requirements for the instrument air system are not impacted by the proposed SPU.

Vents and Drains System

The IST Plan for pumps identifies four floor drain sump pumps in this system. There are no changes to this system as a result of the proposed SPU. The pumps in this system that are in the IST Plan are not changing. Thus, the proposed SPU does not impact the IST Plan requirements for these pumps. Additionally, the IST Plan for valves includes two AOVs and four check valves in this system. The AOVs are the containment isolation valves for the reactor cavity and containment sump discharge heater. No changes are being made to the system as a result of the proposed SPU. The AOVs are designed to meet the containment design pressure of 50 psig. Accident conditions do not exceed containment design pressure. The check valves are discharge check valves for the Safeguard Building floor drain sump 1 and sump 2 discharge lines. No changes are being made to this system as a result of the proposed SPU. Thus, IST Plan requirements for the vents and drains system are not impacted by the proposed SPU.

Miscellaneous Containment Isolation Valves

The IST Plan includes a table listing containment isolation valves that are in IST scope. Tests that measure containment isolation valve leakage rates (Type C tests) are performed at the peak calculated containment internal pressure (Pa) for the design basis loss-of-coolant accident (LOCA). Per the Technical Specifications, the value of Pa is 48.3 psig. From the containment analysis at the SPU conditions, the peak containment pressure is 44.5 psig for a LOCA (see LR

subsection 2.6.1). Since the SPU peak containment pressure is lower than the current peak pressure and is also lower than the Pa, the current test criterion bounds the peak LOCA containment pressure for the proposed SPU. Thus, the IST requirements for the miscellaneous containment isolation valves are not impacted by the proposed SPU. The containment leakage rate test program is discussed below.

Safety and Relief Valves

IST Plan safety and relief valves are located in systems previously discussed. The proposed SPU has no or minimal impact to those systems. Setpoints for these safety and relief valves are not changing. Thus, IST Plan requirements for the safety and relief valves are not impacted by the proposed SPU.

Motor-Operated Valves

The IST Plan identifies motor operated valves. See below for a discussion of the Motor Operated Valve Program and GLs 89-10 and 96-05 evaluations.

Containment Leakage Rate Testing Program

The containment leakage rate testing program addresses leakage rate testing of the containment structure and non-valve penetrations in addition to containment isolation valves. The current Pa in Technical Specifications is greater than the peak LOCA containment pressure for proposed SPU conditions. Thus, the current test pressure criterion is bounding. Technical Specifications and containment leakage rate testing program requirements may be updated later to reflect revised design basis LOCA peak containment pressure for SPU conditions.

Motor-Operated Valve Program

Generic Letters 89-10 and 96-05 MOVs are located in the following systems:

- Auxiliary feedwater
- Component cooling water
- Containment spray
- Chemical and volume control
- Safety injection
- Chilled water
- Residual heat removal
- Service water
- Fire protection
- Reactor coolant

The impact to pressure and flow in these systems has been discussed above with the exception of the fire protection system. The proposed SPU does not impact the fire protection system. As discussed in the auxiliary feedwater subsection above, there is an increase in the minimum flow rate requirement for the turbine-driven auxiliary feedwater pump, but the maximum available

flow rate and pump discharge pressure is not changing. Thus, the MOVs in this part of the auxiliary feedwater system are not impacted. For other systems with IST Plan MOVs, there are no changes in pressure and flow to these systems that include MOVs per the identified GLs. No MOVs are required to be added to the MOV Program as a result of the proposed SPU. The proposed SPU does not impact the MOV Program that is scoped within GLs 89-10 and 96-05. Also, the IST Plan requirements for MOVs are not impacted by the proposed SPU.

Generic Letter 95-07

Pressure locking and thermal binding issue has been previously evaluated. MOV capabilities are adequate. MOV modifications such as valve bonnet pressure reliefs have been implemented as required.

The impact of the SPU results in higher containment sump water temperatures. The previous analysis for pressure locking and thermal binding used conservative temperatures and is still bounding. The proposed SPU is acceptable with regard to pressure locking and thermal binding issues for motor operated valves.

Inservice Inspection Program

The CPNPP Inservice Inspection (ISI) Program provides requirements for testing of all components and piping systems under the jurisdiction of ASME XI. The ISI Program consists of two programs - pressure testing and steam generator tube examination – that implement the necessary requirements. Steam generator tube examination is addressed in LR subsection 2.1.9.

With regard to components and piping systems scoped in the Pressure Testing Program, no changes are being made except for a slight increase in operating system pressure in the feedwater system. The proposed SPU does not add any new components or piping systems. No changes are being made to welds or pipe supports that are addressed by the program. System leakage, functional, and in-service tests are conducted to the requirements of the ASME Code, Section XI. When required, ISI pressure test plans will be adjusted accordingly for this slight change in operating pressure under the proposed SPU conditions.

2.2.4.3 Conclusions

Luminant Power has reviewed the assessments related to the functional performance of safety-related valves (including those valves that provide containment isolation) and pumps and concluded that the effects of the proposed SPU on safety-related valves and pumps have been adequately addressed. Luminant Power further concluded that the effects of the proposed SPU on the MOV Program related to GL 89-10, GL 96-05, and GL 95-07 and on the air-operated program (for high and low safety significant AOVs) have been adequately evaluated and that the lessons learned for those programs have been addressed. Based on this, Luminant Power concluded that safety-related valves and pumps will continue to meet the CPNPP licensing basis with respect to the requirements of GDC-1, GDC-37, GDC-40, GDC-43, GDC-46,

GDC-54, and 10 CFR 50.55a(f) following implementation of the proposed SPU. Therefore, the proposed SPU is acceptable with respect to safety-related valves and pumps.

Luminant Power has reviewed the effects of the proposed SPU on those components and systems scoped within the ISI Program and concludes that these effects have been adequately addressed. Thus, the proposed SPU is acceptable with regard to the ISI Program.

2.2.5 Seismic and Dynamic Qualification of Mechanical and Electrical Equipment

2.2.5.1 Regulatory Evaluation

Mechanical and electrical equipment covered by this section include equipment associated with systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal. Equipment associated with systems essential to preventing significant release of radioactive materials to the environment are also covered by this section. The review for the Comanche Peak Nuclear Power Plant (CPNPP) focuses on the effects of the proposed stretch power uprate (SPU) on the qualification of the equipment to withstand seismic events and the dynamic effects associated pipe whip and jet impingement forces. The primary input motions due to the safe shutdown earthquake (SSE) are not affected by a SPU. The CPNPP acceptance criteria are based on:

- General Design Criterion (GDC) -1, insofar as it requires that structures, systems, and components (SSCs) important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.
- GDC-2, insofar as it requires that SSCs important to safety be designed to withstand the effects of natural phenomena such as earthquakes without loss of capability to perform their safety functions.
- GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accident.
- GDC-14, insofar as it requires that the reactor coolant pressure boundary (RCPB) be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, rapidly propagating fracture and of gross rupture.
- GDC-30, insofar as it requires that components that are part of the RCPB be designed, fabricated, erected, and tested to the highest quality standards practical.
- 10 CFR Part 100, Appendix A, which sets forth the principal seismic and geologic considerations of the seismic and geologic characteristics of the plant site.
- 10 CFR 50, Appendix B, which sets quality assurance requirements for all activities affecting Safety-Related functions of SSCs.

Current Licensing Basis

The CPNPP safety-related SSCs are adequately designed for seismic events relative to conformance to:

- GDC-1 is described in FSAR Section 3.1.1.1, General Design Criteria 1, Quality Standards and Records.
- GDC-2 is described in FSAR Section 3.1.1.2, General Design Criteria 2, Design bases for protection against natural phenomena.
- GDC-4 is described in FSAR Section 3.1.1.4, General Design Criteria 4 – Environmental and Missile Design Bases.
- GDC-14 is described in FSAR Section 3.1.2.5, General Design Criteria 14, Reactor Coolant Pressure Boundary.
- GDC-30 is described in FSAR Section 3.1.4.1, General Design Criteria 30, Quality of Reactor Coolant Pressure Boundary.
- 10 CFR Part 100, Appendix A, Seismic and Geologic Sitting Criteria are described in FSAR Section 2.1.1.
- 10 CFR 50, Appendix B, Quality Assurance 18 Point Criteria are described in FSAR Section 17.2.

2.2.5.2 Technical Evaluation

2.2.5.2.1 Introduction

Safety-related SSCs at CPNPP are designed for both seismic and dynamic events as described in FSAR Sections 3.5, 3.6B, 3.7N and 3.7B, 3.9N and 3.9B, 3.10N and 3.10B, and 3.11N and 3.11B. Missile protection is described in FSAR Section 3.5. Protection against dynamic effects associated with the postulated rupture of piping as discussed in FSAR Section 3.6B. FSAR Sections 3.7N and 3.7B, "Seismic Design," provide the general requirements for seismic design. The requirements for active and passive pumps and valves are described in FSAR Sections 3.9N and B. FSAR Sections 3.10N and 3.10B provide details regarding seismic qualification of safety-related instrumentation and electrical equipment. FSAR Sections 3.11N and 3.11B provide details regarding environmental qualification of safety-related mechanical and electrical equipment.

Seismic requirements are unchanged by the SPU; therefore seismic design is not impacted.

There is no change to seismic inputs (amplified response spectra) or loads resulting from SPU. The existing seismic design basis for piping and supports remains valid and unaffected by SPU.

Hence, piping and seismic support loadings will continue to meet the CPNPP current licensing basis with respect to the requirements of GDC-2.

2.2.5.2.2 Description of Analyses and Evaluations

Seismic input and qualification requirements for safety-related equipment are not affected by SPU. Quality Assurance requirements related to 10 CFR 50, Appendix B are not affected.

Dynamic effects of internally generated missiles under SPU conditions have been evaluated and are addressed in Licensing Report (LR) subsection 2.5.1.2.1, Missile Protection. Dynamic effects of pipe-whip and jet impingement under uprate conditions have been evaluated and are addressed in LR subsection 2.2.1, Pipe Rupture Locations and Associated Dynamic Effects and LR subsection 2.5.1.3, Protection against Postulated Piping Failures in Fluid Systems Outside Containment. The SPU will have no adverse impact on essential equipment as a result of pipe whip, jet impingement and internal missiles.

Evaluations related to dynamic and environmental effects of the SPU are addressed in the following LR sections:

- Piping and supports – Section 2.2.2.2, BOP (All Non-Class 1)
- Nuclear steam supply system (NSSS) piping and supports – Section 2.2.2.1, NSSS – Piping and Supports (Class 1)
- Protection against dynamic effects, including GDC-4 requirements, of missiles, pipe whip and discharging fluids – Section 2.2.1, Pipe Rupture Locations and Associated Dynamic Effects and Section 2.5.1.3, Protection Against Postulated Piping Failures in Fluid Systems Outside Containment.
- Environmental qualification of electrical equipment – Section 2.3.1, Environmental Qualification

2.2.5.3 Conclusion

The evaluation of changes in system design configurations that are required for the proposed SPU concludes that safety-related equipment will continue to be protected from seismic and dynamic events, and will continue to meet the CPNPP current licensing basis. The Luminant Power review of the effects of the proposed SPU on the qualification of mechanical and electrical equipment concludes that the review has:

1. Adequately addressed the effects of the proposed SPU on equipment.
2. Demonstrated that the equipment will continue to meet the CPNPP current licensing basis with respect to the requirements of GDCs-1, -2, -4, -14, -30, and 10 CFR, Part 100, Appendix A; and 10 CFR 50 Appendix B.

2.2.6 NSSS Design Transients

2.2.6.1 Regulatory Evaluation

Luminant Power's review primarily focused on the effects of the proposed stretch power uprate (SPU) on the Comanche Peak Nuclear Power Plant (CPNPP) Units 1 and 2 nuclear steam supply system (NSSS) design parameters upon which the existing applicable NSSS design transients are based, as well as how any differences were reconciled to require that revised NSSS design transients be specified for the operating conditions of the proposed SPU. Luminant Power's acceptance criteria for this review are based on:

- General Design Criterion (GDC) -1, insofar as it relates to components important to safety being designed, fabricated, erected, constructed, tested, and inspected in accordance with the requirements of applicable codes and standards commensurate with the importance of the safety function to be performed.

- GDC-2, insofar as it relates to safety-related mechanical components of systems being designed to withstand seismic events without loss of capability to perform their safety function.

- GDC-14, insofar as it relates to the reactor coolant pressure boundary (RCPB) being designed so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

- GDC-15, insofar as it relates to the mechanical components of the reactor coolant system (RCS) being designed with sufficient margin to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences.

Current Licensing Basis

The adequacy of the CPNPP Units 1 and 2 design relative to the general design criteria is discussed in Final Safety Analysis Report (FSAR) Sections 3.1.1, 3.1.2, 3.1.3, 3.1.4, 3.1.5, and 3.1.6. Specifically, the adequacy of CPNPP Units 1 and 2 design basis design transients was assessed by reviewing conformance to:

- GDC-1 is described in FSAR Section 3.1.1.1, General Design Criteria 1 – Quality Standards and Records. It is noted therein that all structures, systems, and components (SSCs) of the facility were classified according to their importance. The Quality Assurance (QA) Program that is implemented to ensure that safety-related SSCs perform as intended is discussed in FSAR Chapter 17. The QA Program conforms to the intent of 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." Those items for which the requirements of 10 CFR Part 50, Appendix B, are met are listed in the list of quality assured items in FSAR Appendix 17A.

The American National Standards Institute (ANSI) safety classifications are also shown on the flow diagrams presented with their respective sections. Records for the design, fabrication, erection, and testing of safety-related SSCs are maintained as described in FSAR Chapter 17.

- GDC-2 is described in FSAR Section 3.1.1.2, General Design Criteria 2 – Design Bases for Protection Against Natural Phenomena. The design is based on the most severe of the natural phenomena recorded for the site, with an appropriate margin to account for uncertainties in the historical data. The natural phenomena postulated in the design are presented in FSAR Sections 2.3, 2.4, 2.5, and 3.3. The design criteria for the SSCs affected by each natural phenomenon are presented in FSAR Sections 3.2, 3.3, 3.4, 3.5, 3.7, and 3.8.

  Consideration of natural phenomena in the design of SSCs is described in FSAR Sections 3.8, 3.9, and 3.10.

- GDC-14 is described in FSAR Section 3.1.2.5, General Design Criteria 14 – Reactor Coolant Pressure Boundary. The RCPB is designed to accommodate the system pressures and temperatures attained under the expected modes of plant operation, including anticipated transients, and to maintain the stresses within applicable stress limits (FSAR Section 5.2). Also, RCPB materials and selection and fabrication techniques ensure a low probability of gross rupture of significant leakage.

  In addition to the loads imposed on the system under normal operating conditions, consideration is also given to abnormal loading conditions, such as seismic and pipe rupture, which are discussed in FSAR Sections 3.6 and 3.7.

  The leak-before-break methodology demonstrates that the probability of rupturing primary coolant piping is extremely low under design basis conditions. It has been applied to CPNPP Units 1 and 2 to exclude from the design basis the dynamic effects of postulated ruptures in the primary coolant loop piping, as discussed in FSAR Section 3.6B.2.5.1. Implementation of this technology eliminates the need for primary coolant loop piping whip restraints and jet impingement barriers.

  The system is protected from overpressure by means of pressure-relieving devices as required by applicable codes.

- GDC-15 is described in FSAR Section 3.1.2.6, General Design Criteria 15 – Reactor Coolant System Design. The design pressure and temperature for each component in the RCS and associated auxiliary, control, and protection systems are selected to be above the maximum coolant pressure and temperature under all normal and anticipated transient load conditions.

In addition, RCPB components achieve a large margin of safety by the use of proven American Society of Mechanical Engineers (ASME) materials and design codes, use of proven fabrication techniques, nondestructive shop testing, and of integrated hydrostatic testing of assembled components.

FSAR Chapter 5 discusses the RCS design.

2.2.6.2 Technical Evaluation

Input Parameters, Assumptions, and Acceptance Criteria

NSSS design transients are developed for use in the analyses of the cyclic behavior of the NSSS SSCs, and to provide the necessary high degree of integrity for them, the transient parameters selected for component fatigue analyses are based on conservative estimates of the magnitude and frequency of the transients resulting from various plant operating conditions.

As discussed in Chapters 3.9N and 5 of the FSAR, the SSCs important to safety in the RCS and its auxiliary systems are designed to withstand the effects of the cyclic loads from RCS (NSSS) temperature and pressure changes. Such cyclic loading is the result of design basis design transients. The evaluation compared the CPNPP Units 1 and 2 design parameters developed for the SPU (Licensing Report (LR) Section 1.1) to the design parameters used in the existing design basis design transients. Comparative analyses were performed and the transients were revised, as necessary, to reflect the operating conditions for the SPU.

As part of the original design and analysis of NSSS components for CPNPP Units 1 and 2, NSSS design transients (that is, temperature and pressure transients) were specified for use in the analyses of the cyclic behavior of the NSSS components. To provide the necessary high degree of integrity for the NSSS components, the transient parameters selected for component fatigue analyses are based on conservative estimates of the magnitude and frequency of the temperature and pressure transients resulting from various plant operating conditions. The transients selected for use in component fatigue analyses are representative of operating conditions that would be considered to occur during plant operations of possible significance to component cyclic behavior due to their severity or frequency. The selected transients are representative of plant transients that, when used as a basis for component fatigue analysis, would provide confidence that the component is appropriate for its application over the 40-year operating license period of the plant.

The SPU NSSS design transients are based on the NSSS design parameters presented in LR Section 1.1, NSSS Parameters. RCS vessel T_{avg}, T_{hot}, T_{cold}, steam pressure, and feedwater temperature are the critical parameters for design transient considerations.

The original NSSS design transients for CPNPP Unit 1 have recently undergone analyses to support Δ76 steam generator conditions and a revised set of design transients was developed. Therefore, this set of revised design transients for the Δ76 steam generator represents the existing design transients for CPNPP Unit 1. The original NSSS design transients for CPNPP Unit 2 with Model D-5 steam generators are the existing design transients for CPNPP Unit 2.

The design transients were developed assuming that the control and protection systems function as designed, especially for the Condition I transients.

The NSSS design transients are used by the equipment designers as input to the NSSS component structural and fatigue analyses/evaluations. Therefore, there are no acceptance criteria associated with the development of the NSSS design transients. However, the equipment designers determine the final acceptance of the NSSS components in accordance with the appropriate ASME Section III stress and fatigue analyses requirements.

Description of Analyses and Evaluations

The CPNPP Units 1 and 2 design parameters for the existing applicable NSSS design transients were compared to the design parameters for the SPU program. These parameters change for the SPU primarily due to the full-load RCS vessel T_{avg} and feedwater temperature windows. As a result of these parameter changes, evaluations and analyses of the existing applicable NSSS design transients for CPNPP Units 1 and 2 were performed. As necessary, revised NSSS design transients have been specified for the SPU.

The existing NSSS design transients for CPNPP Unit 1 supported an RCS T_{avg} and feedwater temperature window and a NSSS SPU to 3,582 MWt. The existing NSSS design transients for CPNPP Unit 2 reflect a single full-load RCS T_{avg} and feedwater temperature. With the implementation of the T_{avg} and feedwater temperature windows for Unit 2 SPU conditions, the existing design transients for CPNPP Unit 2 required more revisions than CPNPP Unit 1.

Results

NSSS design transients are impacted by the SPU because of changes in plant operating conditions (that is, design condition T_{hot}, T_{cold}, T_{avg}, RCS/pressurizer pressure, steam generator steam pressure, or feedwater temperature). The existing applicable licensing basis transients were reviewed with respect to the SPU conditions and changes to the transients were made for the SPU conditions, as applicable. The differences are primarily due to RCS vessel T_{avg} and feedwater temperature windows for the SPU. In some cases, these differences were sufficient to require a revision of the existing NSSS design transients.

The revised NSSS design transients are specified for the SPU in Table 2.2.6-1. Consistent with the existing NSSS design transients, the revised NSSS design transients are conservative representations of the transients that, when used as a basis for component fatigue analyses, provide confidence that the component remains appropriate for its application over the operating license period of CPNPP Units 1 and 2. These revised transients were used in the NSSS component structural and fatigue evaluations at SPU conditions, LR Section 2.2 Mechanical and Civil Engineering, individually for each component.

A list of the NSSS design transients applicable to CPNPP Units 1 and 2 SPU, with their associated design value frequencies of occurrence, are shown in Table 2.2.6-1. The transients listed and their associated frequencies of occurrence are unchanged from those defined in the existing design basis transient list, as specified in Chapter 3.9N of the FSAR. In addition to the

FSAR transients, additional transients are listed to be consistent with the equipment specifications.

2.2.6.3 Conclusions

The effects of the SPU on the NSSS design transients have been evaluated and the required design transient revisions have been adequately addressed. The revised NSSS design transients have been incorporated into the transient analysis of the safety-related NSSS systems and components and the plant will continue to meet the CPNPP Units 1 and 2 existing licensing basis requirements with respect to GDC-1, -2, -14, and -15 following implementation of the SPU. Therefore, Luminant Power finds the SPU acceptable with respect to the NSSS design transients.

Table 2.2.6-1 List of Design Basis NSSS Design Transients			
Transient Description	**Number of Occurrences**[(1)]	**Transients Required Revision due to the Uprating?**	
		Unit 1	**Unit 2**
Normal Condition Transients			
1. Heatup and Cooldown at 100°F/hr (pressurizer cooldown 200°F/hr)	200	No	No
2. Unit Loading and Unloading at 5%/Minute	13,200	Yes	Yes
3. Step Load Increase and Decrease of 10% of Full Power	2,000	No	Yes
4. Large Step Load Decrease with Steam Dump	200	Yes	Yes
5. Steady-State Fluctuations			
a. Initial fluctuations	1.5×10^5	No	No
b. Random fluctuations	3.0×10^6	No	No
6. Feedwater Cycling at Hot Shutdown	2,000	No	No
7. Not Used	–	–	–
8. Unit Loading and Unloading Between 0% and 15% of Full Power	500	Yes	Yes
9. Boron Concentration Equalization	26,400	No	No
10. Refueling[(2)]	80	No	No
11. Reactor Coolant Pump Startup/Shutdown[(3)]	3,000	No	No
12. Reactor Coolant System Venting[(3)]	320	No	No
Test Conditions			
1. Turbine Roll Test	20	No	No
2. Primary Side Hydrostatic Test	10	No	No
3. Secondary Side Hydrostatic Test	10	No	No
4. Primary Side Leakage Test	200	No	No
5. Secondary Side Leakage Test	80	No	No
6. Tube Leakage Test	800	No	No

Table 2.2.6-1 (cont.) List of Design Basis NSSS Design Transients			
		Transients Required Revision due to the Uprating?	
Transient Description	Number of Occurrences(1)	Unit 1	Unit 2
Upset Condition Transients			
1. Loss of Load, Without Immediate Reactor Trip	80	Yes	Yes
2. Loss of Power (blackout with natural circulation in the reactor coolant system)	40	Yes	Yes
3. Partial Loss of Flow (loss of one pump)	80	Yes	Yes
4. Reactor Trip from Full Power			
a. Without cooldown	230	Yes	Yes
b. With cooldown, without safety injection	160	Yes	Yes
c. With cooldown and safety injection	10	Yes	Yes
5. Inadvertent Reactor Coolant Depressurization	20	Yes	Yes
6. Not used	–	–	–
7. Control Rod Drop	80	No	Yes
8. Inadvertent Emergency Core Cooling System Actuation Injection	60	Yes	Yes
9. Operating Basis Earthquake (20 earthquakes of 10 cycles each)	200	No	No
10. Excessive Feedwater Flow	30	No	No
11. Reactor Coolant System Cold Overpressurization	10	No	No
12. Excessive Preheater Bypass Feedwater Flow(4)	40	N/A	No
13. Split Flow Bypass Valve(4)	40	N/A	No
Emergency Condition Transients			
1. Small Loss of Coolant Accident	5	Yes	Yes
2. Small Steam Line Break	5	No	No
3. Complete Loss of Flow	5	No	Yes

Table 2.2.6-1 (cont.) List of Design Basis NSSS Design Transients			
Transient Description	Number of Occurrences[1]	Transients Required Revision due to the Uprating? Unit 1	Transients Required Revision due to the Uprating? Unit 2
Faulted Condition Transients			
1. Main Reactor Coolant Pipe Break (large loss of coolant accident)	1	No	No
2. Large Steam Line Break	1	No	No
3. Feedwater Line Break	1	No	Yes
4. Reactor Coolant Pump Locked Rotor	1	Yes	Yes
5. Control Rod Ejection	1	Yes	Yes
6. Steam Generator Tube Rupture (included under upset conditions, reactor trip from full power with safety injection)	1	No	No
7. Safe Shutdown Earthquake	1	No	No

Notes:

1. Number of occurrences remains unchanged from the existing design basis.
2. Applicable to the RCP and reactor vessel components.
3. The RCP startup/shutdown and RCS venting transients are not defined in the FSAR. However, these transients are included in the existing Unit 1 Model Δ76 and Unit 2 Model D-5 steam generators design specifications.
4. The excessive preheater bypass feedwater flow and split flow bypass valve transients are not described in the FSAR. However, these transients are included in the existing Unit 2 Model D-5 steam generators design specifications. Unit 1 does not have a preheater bypass or split flow bypass line.

2.2.7 Bottom-Mounted Instrumentation Guide Tubes and Flux Thimbles

2.2.7.1 Regulatory Evaluation

The bottom-mounted instrumentation (BMI), along with the in-core thermocouples, comprise the reactor in-core instrumentation system that is designed to yield information on the neutron axial flux distribution and fuel assembly outlet temperatures at selected core locations. This system is employed to evaluate the core power distributions throughout core lifetime to ensure that the thermal design criteria are met. The system provides means for acquiring data, and performs no operational plant control.

The Luminant Power review of the BMI system focused on the effects of the proposed stretch power uprate (SPU) on the structural integrity of the components of the system and its continued functionality, including the capability to maintain the reactor coolant system (RCS) pressure boundary, and withstand any adverse dynamic loads under the maximum temperatures and pressures associated with the proposed SPU. The Luminant Power acceptance criteria for the BMI components are based on:

- General Design Criterion (GDC) -1 and 10 CFR 50.55a, insofar as they require that structures, systems, and components (SSCs) important to safety be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance the safety functions to be performed.
- GDC-2, insofar as it requires that SSCs important to safety be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions.
- GDC-4, insofar as it requires that safety-related SSCs be designed to accommodate and be compatible with specified environmental conditions, and be appropriately protected against dynamic effects, including the effects of missiles.
- GDC-14, insofar as it requires that the reactor coolant pressure boundary (RCPB) be designed, fabricated, erected and tested so as to have an extremely low probability of rapidly propagating fracture.

Current Licensing Basis

GDC-1 – quality standards and records, is described in Final Safety Analysis Report (FSAR) Section 3.1.1.1.

Structures are identified and classified in accordance with the requirement that they be designed to withstand the effects of earthquakes, as delineated in FSAR Section 3.2.

The systems and components of the facility are classified according to their importance in the prevention and mitigation of accidents. Reactor components use the classification system developed by American National Standards Institute (ANSI) N18.2-1973, Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants.

The codes, standards, and quality control applicable to each type of component are identified in pertinent equipment specifications. Where applicable, design and fabrication are in accordance with the codes specified in 10 CFR Part 50, Section 55a. See FSAR Section 5.2.1.1. Alternative requirements, as provided by ASME Code Cases, are utilized at the Comanche Peak Nuclear Power Plant (CPNPP) in accordance with 10 CFR Part 50, Section 55a(a)(3). By reference to ASME Section III requirements in the procurement specifications, the use of code cases by mechanical equipment suppliers requires mutual consent of the Owner or his agent and the manufacturer.

GDC-2 – design bases for protection against natural phenomena, is described in FSAR Section 3.1.1.2.

The natural phenomena and their magnitude are selected in accordance with their probability of occurrence at the CPNPP site. The design is based on the most severe of the natural phenomena recorded for the site, with an appropriate margin to account for uncertainties in the historical data. The natural phenomena postulated in the design are presented in FSAR Sections 2.3, 2.4, 2.5, and 3.3. The design criteria for the structures, systems, and components affected by each natural phenomenon are presented in FSAR Sections 3.2, 3.3, 3.4, 3.5, 3.7, and 3.8.

Combinations of natural phenomena and plant originated accidents considered in the design are identified in FSAR Sections 3.8, 3.9, and 3.10. The importance of the safety functions is identified with the classification system developed by the American Nuclear Society (ANS) and is generally in accordance with ANSI N18.2-1973, Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants. This identification and any deviations are included in FSAR Section 3.2.

GDC-4 – environmental and dynamic effects design bases, is described in FSAR Section 3.1.1.4.

The station's SSCs important to safety are designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including a loss-of-coolant accident (LOCA). Environmental conditions are described in FSAR Section 3.11.

These SSCs are appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that can result from equipment failures and from events and conditions outside the nuclear power unit.

Details of the design, environmental testing, and construction of these systems, structures, and components are included in FSAR Chapters 3, 5, 6, 7, 8, 9, and 10. Evaluation of the performance of safety features is contained in FSAR Chapter 15.

The leak-before-break methodology demonstrates that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping. It has been applied to CPNPP Units 1 and 2 to exclude from the design basis the dynamic effects of

postulated pipe ruptures in the primary coolant loop piping and 10 inch and larger reactor coolant loop branch lines, as discussed in FSAR Sections 3.6B.2.5.1 and 3.6B.2.5. Implementation of this technology eliminates the need for pipe whip restraints and jet impingement barriers, respectively. Containment design, emergency core cooling, and environmental qualification requirements are not influenced by this modification.

GDC-14 – reactor coolant pressure boundary, is described in FSAR Section 3.1.2.5.

The RCS pressure boundary is designed to accommodate the system pressures and temperatures attained under the expected modes of plant operation, including anticipated transients, and to maintain the stresses within applicable stress limits (FSAR Section 5.2). Also, RCPB materials and selection and fabrication techniques ensure a low probability of gross rupture of significant leakage.

In addition to the loads imposed on the system under normal operating conditions, consideration is also given to abnormal loading conditions, such as seismic and pipe rupture, which are discussed in FSAR Sections 3.6 and 3.7.

The leak-before-break methodology demonstrates that the probability of rupturing primary coolant piping is extremely low under design basis conditions. It has been applied to CPNPP Units 1 and 2 to exclude from the design basis the dynamic effects of postulated ruptures in the primary coolant loop piping, as discussed in FSAR Section 3.6B.2.5.1. Implementation of this technology eliminates the need for primary coolant loop piping whip restraints and jet impingement barriers. Containment design, emergency core cooling, and environmental qualification requirements are not influenced by this modification.

In conclusion, the RCS boundary has provisions for inspection, testing, and surveillance of critical areas to assess the structural and leaktight integrity (FSAR Section 5.2). For the reactor vessel, a material surveillance program conforming to applicable codes is provided in FSAR Section 5.3.

2.2.7.2 Technical Evaluation

Input Parameters, Assumptions, and Acceptance Criteria

The BMI system consists of the reactor vessel, retractable flux thimbles, miniature flux detectors, guide tubing, and the seal table. The retractable flux thimbles into which the detectors are driven enter the guide tubing at the seal table, pass through the tubing into the reactor vessel, through the lower internals instrument columns, and then into the fuel. Each detector provides axial flux distribution data along the center portion of a fuel assembly. This data is then processed to obtain a core flux map.

The leading ends of the flux thimbles are closed and bullet-nosed. The thimbles are dry inside, and serve as a pressure barrier between the RCS and the containment atmosphere. The pressure boundary for the guide tubing and flux thimbles is maintained by the compression fittings at the seal table where one end of each guide tube has a compressed fitting connection.

The other end of each guide tube is welded to the bottom penetration nozzle of the reactor vessel bottom head.

The guide tubing material is ASME SA213 Type 304L stainless steel, cold drawn and heat treated. The tubes are pressure boundary components designed to ASME Boiler and Pressure Vessel Code Section III (Reference 1), Code Class 1, subsection NC specifications. The flux thimble material is ASME SA213 Type 316, cold drawn and heat treated. The seal table is a rectangular plate designed to ASME Section III, Class 1, subsection NF requirements. The reactor BMI system guide tubing is classified as Seismic Category I. The flux thimble for the BMI system is classified as Seismic Category II, since it is an instrument tube.

The analyses and evaluation of the CPNPP BMI system presented herein are an assessment of the impact on the structural integrity of the system from the thermal transients and maximum operating temperatures, pressures, and design basis accident (DBA) displacement that result from the proposed uprate operating conditions. The results of these analyses and evaluations showed that stresses in the BMI guide tubing remained within the allowable limit.

The BMI guide tubing is designed to meet the ASME Boiler and Pressure Vessel Code, Section III (Reference 1), Class 1 criteria. However, per Subsection NB-3630, Par. D.1 of the Code, "Piping of 1 inch nominal pipe size or less which has been classified as Class 1 in the design specification may be designed in accordance with the design requirements of Subsection NC." The flux thimble is classified as an instrument tube, so it is beyond the jurisdiction of the ASME Code per NA-1130(c). The flux thimbles were qualified as part of BMI guide tubing. No separate qualification of flux thimble was needed. The weight of the flux thimble was considered in the qualification of BMI guide tubing.

The input parameters were:

- Operating conditions (LR Section 1.1)
- NSSS design transients (LR subsection 2.2.6)
- LOCA displacement information at the bottom of reactor vessel head

Description of Analyses and Evaluations

There are three areas of interest for the reconciliation of BMI guide tubing qualification. They are:

1. Pressure increase during transients

2. Temperature increase during transients, and the revised core inlet temperature from LR Section 1.1 of this report

3. Reactor vessel bottom dome displacement during a LOCA

The BMI guide tubing is qualified for 2,485 psig, and a reactor coolant temperature of 650°F for Unit 1 and 680°F for Unit 2. The temperature of the guide tube at the seal table is ambient

(60-120°F during normal conditions). The temperature in the BMI guide tubing is attenuated from the reactor pressure vessel end to the seal table. The temperature and pressure values for the proposed SPU are compared with the qualified values, and the calculated stress in the guide tubing is adjusted as a result of increased temperature and/or stress. Likewise, the reactor vessel displacement at the bottom dome of the reactor vessel during a postulated LOCA for the proposed SPU is compared with the original equipment qualification, and the calculated stress in the guide tubing is adjusted as a result of greater displacement. Since the seismic loading during uprating remained unchanged, the stresses due to operating basis earthquake (OBE) and safe shutdown earthquake (SSE) seismic loadings remain unchanged.

Equations 8 through 11 from ASME III (Reference 1) were re-evaluated for the above three changes.

None of the pressure transients identified in the NSSS Design Transients were revised for the SPU. Therefore, it was not necessary to calculate a new pressure stress value. Some T_{cold} temperature transients were revised as a result of the SPU. For Unit 1, the revised maximum temperature during a transient (606°F) does not exceed the previous qualified temperature of 650°F. For Unit 2, a new temperature stress value is calculated due to a revised maximum temperature during a transient (690.5°F) which exceeds the previous qualified temperature of 680°F by 1.5 percent. The stresses due to seismic and LOCA are combined using the square root sum of the squares (SRSS) method. The Unit 1 faulted condition stress is determined to be below the allowable stress based on a comparison of Unit 2 stress values. For Unit 2, the equation 9 faulted condition stress was a maximum of 38 percent higher than the equation 9 upset condition stress. The allowable stress for the faulted condition is 2 times the allowable stress for the upset condition. Since the Unit 1 equation 9 upset stress is below the allowable stress, it follows that the equation 9 faulted stress is below the allowable stress. Furthermore, the displacements of the vessel bottom head as a result of a LOCA used for Unit 2 apply to Unit 1. For Unit 2, the new displacement values are bounded by the displacement values of the initial qualification evaluation, thus the stresses due to the displacements are also bounded by the initial site evaluation. The revised total stress values of equations 8 through 11 are then compared with the respective allowable stress value for each condition.

Note: The flux thimbles were included in the guide tubing qualification by incorporating the flux thimble mass value with the guide tube mass value. There was no separate qualification for the flux thimble, since it is classified as an instrument tube.

Results

As shown in Tables 2.2.7-1 and 2.2.7-2, the results of analyses and evaluations of the proposed SPU operating conditions are that stresses in the BMI guide tubing remain within the allowable limit. The minimum stress margin for Unit 1 is 13.4 percent, and the minimum stress margin for Unit 2 is 4.3 percent.

2.2.7.3 Conclusions

Luminant Power has reviewed the assessment of the effects of the SPU on the in-core BMI system and has determined that it has adequately accounted for the effects of changes in plant conditions on the design of the BMI system. Luminant Power concludes that the BMI system will maintain its structural integrity under the operating conditions of the proposed SPU. Luminant Power further concludes that the BMI system will continue to meet the CPNPP current licensing basis requirements with respect to GDC-1, -2, -4, and -14.

2.2.7.4 References

1. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III, 1974 Edition Including Summer 1974 Addenda.

Table 2.2.7-1 Unit 1 Stress Summary of ASME III Equations 8 through 11			
Equation No.	**Stress (psi)**	**Allowable Stress (psi)**	**Minimum Stress Margin (%)**
8 (Design)	5,210	15,900	67.2
9 (Normal and Upset)	16,528	19,080	13.4
9 (Emergency)	21,325	28,620	25.5
10 (Normal and Upset)	14,509	27,475	47.2
11 (Normal and Upset)	19,189	43,375	55.8

Table 2.2.7-2 Unit 2 Stress Summary of ASME III Equations 8 Through 11							
Equation No.	**Stress Short Tube (psi)**	**Stress Medium Tube (psi)**	**Stress Long Tube (psi)**	**Stress Medium Tube with Tee (psi)**	**Stress Long Tube with Tee (psi)**	**Allowable Stress (psi)**	**Minimum Stress Margin (%)**
8 (Design)	3,853	4,541	4,854	7,379	5,793	15,900	53.6
9 (Upset)	6,620	7,584	8,410	12,244	10,271	19,080	35.8
9 (Faulted)	9,130	9,879	10,564	15,653	12,957	38,160	59.0
10 (Normal)	21,067	11,457	13,079	11,456	13,078	27,475	23.3
10 (Upset)	42,132	18,509	20,020	18,508	20,019	27,475	-53.3[(1)]
11 (Normal)	24,102	14,151	15,658	14,148	15,658	43,375	44.4
11 (Upset)	41,513	21,837	22,600	21,829	22,600	43,375	4.3

Note:

1. Per ASME Code, Paragraph NC-3650, the requirements of either equation 10 or equation 11 must be met.

2.3 ELECTRICAL ENGINEERING

2.3.1 Environmental Qualification of Electrical Equipment

2.3.1.1 Regulatory Evaluation

Environmental qualification (EQ) of safety-related electrical equipment involves demonstrating that the equipment is capable of performing its safety function under significant environmental stresses which could result from design basis accidents (DBAs). The review focused on the effects of the proposed stretch power uprate (SPU) on the environmental conditions that the electrical equipment will be exposed to during normal operation, anticipated operational occurrences, and accidents. The review was conducted to ensure that the electrical equipment will continue to be capable of performing its safety functions following implementation of the proposed SPU. The acceptance criteria for EQ of electrical equipment are based on 10 CFR 50.49, which sets forth requirements for the qualification of electrical equipment important to safety that is located in a harsh environment.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Additional details that define the licensing basis for safety-related electrical equipment qualification are described in the FSAR Section 3.11. The Equipment Qualification Report has been submitted to the NRC as Appendix 3A as referenced in the FSAR. This appendix covers the station operations and maintenance Equipment Qualification Program. The Equipment Qualification Program was audited by the NRC and determined to be acceptable. Discussion regarding the tests and analyses used to demonstrate the seismic design of safety-related electrical equipment is provided in FSAR Sections 3.7, 3.9, and 3.10. The primary input motions attributable to an earthquake are not impacted by the proposed SPU.

The EQ Program complies with 10 CFR Part 50.49. The EQ Program ensures the continued qualification of safety-related electrical equipment that must function during and following design conditions postulated for DBAs and the post-accident duration.

The NRC required that CPNPP perform a review and evaluation of EQ of safety-related electrical equipment in potentially harsh environments at CPNPP. CPNPP Units 1 and 2 and are committed to maintain the EQ Program.

The NRC Safety Evaluation Report concluded that the CPNPP Equipment Qualification Program complied with 10 CFR 50.49, GDC 1, 4 and Appendix B to 10 CFR 50 requirements.

2.3.1.2 Technical Evaluation

2.3.1.2.1 Introduction

Safety-related structures, systems and components (SSCs) at CPNPP are designed for environmental events as described in FSAR Sections 3.10 and 3.11. FSAR Section 3.10 provides the details regarding seismic qualification of safety-related mechanical, structural, instrumentation and electrical equipment. FSAR Section 3.11 provides details regarding environmental qualification of safety-related electrical equipment.

The constituent parts of the EQ Program include the program basis, verification of equipment operability during and following exposure to plant environmental conditions, and proper installation and maintenance of equipment in the plant. These elements are controlled through a set of administrative documents consisting of a program description, implementing procedures, and reference documents.

Program output is documented in Electrical Equipment Qualification Summary Packages. These documents provide the auditable bases and evidence, which demonstrates that CPNPP is compliant with 10 CFR 50.49 and GDC 1 and 4.

An evaluation of the environmental qualification of equipment in the Equipment Qualification Program demonstrates that all equipment will remain qualified under SPU conditions, but with changes to the conditions under which it is qualified. The qualification evaluation reviewed the changes to equipment environments resulting from anticipated operational occurrences, normal operation, accidents, and post-accident conditions.

Comparisons were made between the environmental conditions to which the equipment is currently qualified and the environmental conditions that will be present following the implementation of the SPU. Where there is a qualification challenge, further evaluations specific for the equipment will be performed and where necessary equipment updated.

The evaluations demonstrated the continued qualification of the equipment or identified the requirement for equipment upgrades or changes by ensuring that the margins required by the Institute of Electrical and Electronic Engineers (IEEE)-323 and the CPNPP Equipment Qualification Program are maintained. The acceptance criterion is the continued qualification of the equipment under the requirements of 10 CFR 50.49.

2.3.1.2.2 Description of Analysis and Evaluations

Introduction

The current normal operation dose estimates utilized for equipment qualification are based on either the normal operation radiation zone dose rate limit, or on radiation source terms corresponding to a core power level of 3,565 MWt and 1 percent fuel defects. With the exception of a few selected components, integrated doses are based on 40 years of normal operation. Inside containment, the current normal operation dose contribution reflects an

80-percent plant capacity factor; outside containment, the capacity factor is assumed to be 100-percent.

The accident doses are currently based on a core power level of 3,565 MWt, and with the exception of a few selected components, an integration period of one year. The accidents that control post accident environments include the loss-of-coolant accident, the high energy line break and the fuel handling accident.

The SPU will typically increase the radioactivity level in the core by the percentage of the uprate. The radiation source terms in equipment/structures containing radioactive fluids, and the corresponding radiation zone doses, will increase accordingly. Additional factors that impact the equilibrium core inventory and consequently, the estimated dose, are fuel enrichment and burnup.

The impact of SPU on the environmental dose estimates is developed using scaling techniques that reflect a comparison of the source terms developed based on the core inventory utilized in the analyses of record to the corresponding source terms developed using the SPU core inventory. Since the relative abundance of each isotope and the average gamma energy of each isotope are the key parameters that affect direct exposure, having a scaling factor that addresses the change in these parameters is sufficient to assess the radiological impact of the SPU. Consideration is given to the fact that dose scaling factors will vary with radiation source, time after accident, as well as shielding, therefore bounding values are utilized.

The SPU core inventory is based on a conservative core power level of 3,684 MWt and an 18-month fuel cycle. In addition to the SPU power level, the estimated increase in radiation levels reflects the use of:

1. The ORIGEN-S computer code, and associated nuclear data libraries in developing the SPU core inventory (vs. the original ORIGEN code and libraries used in the development of the original licensing basis core inventory). See Appendix A, Codes and Methods; and

2. An additional 4-percent margin utilized in development of the SPU core to address potential future fuel cycle variability.

As a result, the calculated SPU dose scaling factor values are different from the SPU ratio. Note that the impact of the change in the analyzed power level (3,565 MWt to 3,684 MWt) and the 4 percent margin alone account for a 7.5 percent increase.

Inside containment, the scaling factor increase for the normal operation dose contribution also includes an assumed increase in plant capacity factor from 0.8 to 1.0. Outside containment, for areas where the fuel handling accident is controlling, the calculated post accident contribution to the SPU dose is based on fuel movement assumed to occur 75 hours post shutdown vs. the current analyses that reflect fuel movement at T=100 hours after shutdown.

Inside Containment

With the exception of radiation and temperature, normal service conditions and operational occurrences (that is, for pressure and humidity) do not change. The radiation environments are increased to reflect the SPU, and to acknowledge an increase in plant capacity factor. The normal containment temperature is increased by less than 1°F, however remains below the maximum allowable temperature of 120°F. Refer to Licensing Report (LR) subsection 2.7.7.

The SPU containment analysis (refer to LR subsection 2.6.1) of DBAs demonstrates that the equipment qualification peak temperature and pressure is bounded by the current EQ profile. However, the long-term pressure and temperature slightly exceed the current profile at limited times later in the transient after the peak pressure and temperature have been reduced. Pressure effects are generally stress-related rather than age degradation related. Qualification to the high-pressure peak ensures that there are no stress related component failures. For the temperature impact, the post-accident operating time has been evaluated and found acceptable.

The radiation doses inside containment are increased due to the SPU. Revised total integrated doses in each environmental zone that combine normal operation service conditions with the DBAs have increased up to a maximum of 11 percent and have been compared to the original qualification value. The reactor cavity area integrated dose increased by approximately 22 percent due to the assumed increase in plant capacity factor from 0.8 to 1.0. The results of the comparison shows that all the Equipment Qualification Program electrical equipment will continue to be qualified at the SPU conditions and thus will continue to meet the requirements of 10 CFR 50.49. Where the increase in radiation exceeds the current EQ limits, additional analysis will be performed to document that the affected components specific dose is bounded by the specific components EQ qualification.

Other parameters that affect the qualification of equipment are humidity, submergence, and chemical spray. These will not change due to SPU.

Outside Containment

Normal service conditions and operational occurrences for, temperature, pressure, and humidity do not change following implementation of the SPU. The normal radiation dose for some rooms and cubicles will increase slightly to account for the SPU.

The accident environments outside containment have been evaluated. There is a small temperature increase from existing HELB temperatures in the main steam and feedwater penetration areas. The impact on EQ components has been evaluated to demonstrate that the conditions for which the components have been qualified are acceptable. Where the increase in temperature exceeds the current EQ limits, additional analysis will be performed to document that the affected components specific qualification bounds the affect of the temperature increase.

For areas outside containment with harsh radiation environments, the total integrated doses to the equipment have been scaled up, where applicable, to account for increases in normal operating and post-accident radiation levels as a result of the SPU.

The radiation doses outside containment are also increased due to the SPU. The increases in the total integrated doses in each environmental zone due to the SPU (except as noted below) were estimated at a maximum increase of 8 percent and then compared to the EQ doses. The integrated dose near the fuel pool is estimated to increase by 13 percent primarily due to a conservative early fuel off load at 75 hours. The results of the comparison shows that all the Equipment Qualification Program electrical equipment will continue to be qualified at the SPU conditions and thus will continue to meet the requirements of 10 CFR 50.49.

2.3.1.3 Conclusion

The electrical equipment has been evaluated for SPU environmental conditions during normal operation, operational occurrences, and following DBAs. The SPU evaluation concluded that equipment will be capable of performing its safety function under the environmental conditions that could result from design basis accidents. The evaluation indicated that the electrical equipment in the Equipment Qualification Program continues to be qualified in accordance with 10 CFR 50.49.

The proposed SPU does not impact the input criteria used for the seismic design and qualification of CPNPP electrical equipment. Consequently, this equipment will continue to meet the requirements of GDC 1 and 4 following the implementation of the proposed SPU.

2.3.2 Offsite Power System

2.3.2.1 Regulatory Evaluation

The offsite power system includes independent circuits capable of operating independently of the onsite standby power sources. The Comanche Peak Nuclear Power Plant (CPNPP) review covered the information, analyses, and referenced documents for the offsite power system. The acceptance criteria for the offsite power system are based on General Design Criterion (GDC) -17, insofar as it requires the system to have the capacity and capability to perform its intended functions during anticipated operational occurrences and accident conditions.

Current Licensing Basis

Final Safety Analysis Report (FSAR) Section 8.2.2 states that the offsite power system is designed with sufficient capacity and independence to ensure performance of their safety functions.

FSAR Sections 8.2.1.1 and 8.2.1.2 demonstrates that the offsite power system comply with the design requirements of 10 CFR 50, Appendix A, GDCs-17 and -18 and Nuclear Regulatory Commission (NRC) Regulatory Guide 1.32. The overall system design including functional requirements, independence, capacity, and availability is in conformance with Institute of Electrical and Electronics Engineers (IEEE) 308-1974.

2.3.2.2 Technical Evaluation

2.3.2.2.1 Introduction

As described in FSAR Sections 8.1.1 and 8.2.1, the offsite power system has the following key design attributes.

The CPNPP output is connected to the 345 kV system via the transmission service provider's CPNPP 345 kV switchyard. Each unit output line is connected to both East and West 345 kV busses through their respective 345 kV circuit breakers in the switchyard.

The Electric Reliability Council of Texas (ERCOT) assures the stability and adequacy of CPNPP offsite power sources.

The CPNPP switchyard consists of the 345 kV and the 138 kV switchyards, and each includes a two bus arrangement having one breaker per transmission circuit. Transmission circuits terminate in individual positions on alternate busses in the switchyards. The CPNPP output is connected to the 345 kV transmission system via the 345 kV switchyard. The 345 kV switchyard provides the normal power for Unit 1 safeguard busses and the alternate power for Unit 2 safeguard busses. The 138 kV switchyard provides the normal power for Unit 2 safeguard busses and the alternate power for Unit 1 safeguard busses.

The offsite power source for startup transformer XST2, spare startup transformer XST1/2, and station service transformer 1ST are connected to a common overhead line from 345 kV switchyard. This overhead line is connected to both East and West 345 kV busses through circuit breakers 7970 (E6) and 7980 (W6). Transformers XST2 and 1ST are provided with a 345 kV motor-operated air switch such that each transformer can be energized independent of the other transformer.

Station service transformer 2ST is connected to the 345 kV switchyard west bus via a dedicated overhead line and circuit breaker 8080 (W11).

The CPNPP offsite power source for start up transformer XST1 is connected to both 138 kV busses through circuit breakers 7030 and 7040 and motor-operated air switch DXST1(8085).

The existing protective system relay settings for the startup and station service transformers are based on maximum MVA ratings. The tie-lines connecting these transformers to the switchyard are also protected by these devices. The protective relaying system is adequate and no changes need to take place for stretch power uprate (SPU).

Current protective system relay settings for the Units 1 and 2 main transformers are based on maximum MVA ratings. The tie-lines connecting these transformers to the switchyard are also protected by these devices. Due to SPU increased generation capability there will be an increase in current flowing through the Units 1 and 2 main transformers and the tie-lines connecting the units to the switchyard. The existing protective system relay settings will be adjusted as required to reflect the change in the equipment size, and increase in the load flow in the tie lines connecting the Units 1 and 2 main transformers to the switchyard. The control of the switchyard is administered through an interconnect agreement between Luminant Power and the transmission service provider.

2.3.2.2.2 Description of Analyses and Evaluations

The offsite power system and its components were evaluated to ensure they are capable of performing the system's intended function at SPU conditions. The evaluation is based on the system's required design functions and attributes, and upon a comparison between the existing offsite power system equipment ratings and the anticipated operating requirements at SPU conditions.

FSAR Section 8.2 evaluates offsite power system and states that they comply with the intent of 10 CFR 50, Appendix A GDC-17 and -18, Regulatory Guides 1.30, 1.32, 1.47 and 1.93, and IEEE 308-1974.

2.3.2.2.3 Results

2.3.2.2.3.1 Grid Stability and Switchyard Components

The offsite power system is discussed in FSAR Sections 8.1.1 and 8.2.1. ERCOT, through the transmission service provider, has evaluated steady-state and stability studies for the impact of the SPU on the reliability of the CPNPP 345 kV switchyard. The transmission service provider has concluded that the steady-state and dynamic performance of the CPNPP at SPU is essentially unchanged and therefore acceptable. The short circuit levels remain acceptable for the CPNPP switchyard equipment ratings and transmission circuit line capacities were sufficient to handle the proposed increase in generation capability at CPNPP without any required hardware modifications. CPNPP VARs (volt-amperes reactive) support of the grid continues to be adequate.

The above described evaluation of system impacts demonstrates that the steady-state and dynamic performance of the CPNPP at SPU conditions is acceptable and offsite power system design continue to meet the intent of GDC-17 at SPU conditions.

The 138 and 345 kV switchyard and distribution system were evaluated to ensure required functions are performed after the implementation of SPU and, consequently, to ensure the functionality of the switchyard and its associated components. The evaluation determined that no changes are required to the 138 or 345 kV switchyard equipment or associated components. The 138 and 345 kV switchyard equipment ratings, including the tie lines to the high voltage bushings of the main transformers, startup transformer, or station services transformers were determined to bound the SPU requirements. The tie-lines conductor clearances at maximum ambient temperature and maximum SPU current meet industry requirements.

Two separate and physically independent startup transformers provide startup, preferred and alternate shutdown power to the safety-related auxiliaries on an immediate basis. One transformer is connected to the 345 kV switchyard while the second transformer is connected to the 138 kV switchyard. These transformers are connected to the safety-related bus system of each unit and, as such, provide two independent means of supplying the safety-related equipment from the offsite power.

The switchyard configuration has not changed due to SPU, and it continues to provide a reliable offsite power supply path to the station onsite distribution system through the main transformers, startup transformer, and station services transformers. The separate circuits satisfy the independence and redundancy requirements of GDC-17.

2.3.2.3 Conclusion

As a result of the evaluations for SPU, it has been determined that the offsite power system steady-state and dynamic performance at SPU is acceptable. The short circuit levels remain acceptable for the CPNPP switchyard equipment ratings and transmission circuit line capacities were sufficient to handle the proposed increase in generation capability at CPNPP without any required hardware modifications. Adequate physical and electrical separation and equipment

margin exists and the offsite power system has the capacity and capability to supply power to all safety loads and other required equipment at SPU.

The evaluation concluded that the Units 1 and 2 offsite power systems will continue to function as designed and continue to meet the current licensing basis with respect to the requirements of 10 CFR 50 Appendix A, GDC-17 following implementation of the proposed SPU.

2.3.3 AC Onsite Power System

2.3.3.1 Regulatory Evaluation

The alternating current (AC) onsite power system includes those standby power sources, distribution systems, and auxiliary supporting systems provided to supply power to safety-related equipment. The AC onsite power system for Comanche Peak Nuclear Power Plant (CPNPP) Units 1 and 2 consists of the main generators, isolated phase bus ducts, main transformers, unit auxiliary transformers, station service transformers and startup transformers, AC distribution medium voltage (6.9 kV and 2.4 kV), low voltage (480V, 277, 240, 208,120V and 118V), and standby power emergency diesel generator systems.

The acceptance criteria for the AC onsite power system are based on General Design Criterion (GDC)-17, insofar as it requires the system to have the capacity and capability to perform its functions during anticipated operational occurrences and accident conditions.

Current Licensing Basis

Final Safety Analysis Report (FSAR) Section 8.3.1.2.1 notes that the AC onsite power systems are designed with sufficient capacity, independency, and redundancy to ensure performance of their safety functions assuming a single failure.

As stated in FSAR Sections 3.1.2.8 and 8.3.1.2.1, the AC onsite power system is in conformance with the GDC-17 for nuclear power plants.

Additional details that define the licensing basis are described in FSAR Sections 8.1.2, Onsite Electric System Description; and 8.3.1, AC Power System.

2.3.3.2 Technical Evaluation

2.3.3.2.1 Introduction

The AC onsite power system and components are discussed in FSAR Sections 8.1.2 and 8.3.1. The AC onsite power system consists of:

- Main generator for each unit is rated 1,410 MVA, 22KV, 60 HZ, 0.9 power factor, 1,800 RPM @ 75 psig hydrogen pressure.

- Isolated phase bus duct conducts electrical power from the main generator to the main transformers and unit auxiliary transformer.

- Main transformers are provided to raise the generator voltage from 22 to 345 kV switchyard voltage.

- Unit auxiliary transformer supplies power for non-Class 1E auxiliaries during normal operation from 22 to 6.9 kV for the non-Class 1E 6.9 kV busses.

- Station service transformer for each unit step down switchyard voltage from 345 kV to 6.9 kV for the supply of power to the non-Class 1E 6.9 kV busses.

- Startup transformers supplies power to both Unit 1 and Unit 2 from the 138 kV and 345 kV switchyards to the safety-related (Class 1E) 6.9 kV busses.

- The medium voltage 6.9 kV AC system busses supply power for operation of Class 1E and non-Class 1E 6.9 kV loads.

- The low voltage 480V system (Class 1E and non-Class 1E) are powered through step down 6.9 kV/480V load center transformers.

- The low voltage AC system consists of normal 480/277V, 208/120V, 118V and 240/120V AC subsystem, and Class 1E 208/120V, 120V and 118V AC subsystems. The 118V AC subsystem is the uninterruptible power supply system that provides continuous power to all instrument and control equipment.

- The standby power source consists of two emergency diesel generators per unit and provides power to Class 1E loads.

2.3.3.2.2 Description of Analyses and Evaluations

The AC onsite power system and components were evaluated to ensure they are capable of performing the intended systems function at stretch power uprate (SPU) conditions. The evaluation was based on the required system design functions and attributes, and upon a comparison between the existing equipment ratings and the anticipated operating requirements at SPU conditions.

2.3.3.2.3 Results

The main generator capability curve has been revised based on a Siemens generator uprate study. The study indicates that the new uprate main generator nameplate rating of 1,410 MVA @ 0.9 power factor is adequate to support unit operation at SPU conditions. Modifications are required to enhance cooling to support SPU conditions. The cooling modifications are the replacement of the main generator hydrogen coolers, replacement of the iso-phase coolers and fans, and additional cooling for the exciter cooler.

Evaluation of the main generator protection confirms that the main generator total and partial loss of field and negative sequence relays settings are affected by the SPU conditions. The settings for these relays will be adjusted to support the SPU. The remaining main generator protective relaying schemes and setpoints are not affected as a result of SPU conditions.

The evaluation also demonstrates that the isolated phase bus duct main and tap bus short circuit design ratings envelope the available fault current levels for SPU conditions. The evaluation also confirms that the existing isolated phase bus duct main generator and main

transformer tap busses continuous current design ratings are inadequate to support unit operation at SPU conditions. Modifications will be implemented to support SPU conditions.

The evaluation confirms that the current main transformers, with existing administrative limits, are adequate for the SPU. The MVA capacity supports the SPU output of the main generators and the MVAR commitment to the transmission provider.

The existing main transformer protective relaying schemes and settings are not affected by SPU conditions.

The evaluation confirms that the unit auxiliary transformers and station service transformers design ratings for each unit are adequate to support unit operation at the SPU.

The evaluation confirms that there are no load changes to the startup transformers due to SPU conditions. The startup transformers design ratings remain bounded at SPU conditions.

The evaluation of the 6.9 kV load flow and short circuit analyses determined that the 6.9 kV non-segregated phase bus duct is adequate to support SPU conditions.

The evaluation confirms that the existing non-Class 1E 6.9 kV bus continuous current design ratings are adequate to support unit operation at SPU conditions. SPU load changes to the 6.9 kV busses have been analyzed. It has been determined that the bus loading and short circuit current are not exceeded and confirmed that the expected performance is met at SPU conditions.

The evaluation of the existing protection system relay settings for the non-Class 1E 6.6 kV rated reactor coolant pump (RCP) motors confirms that the applied protective relaying schemes and setpoints for RCP hot and cold loop motor operation and reactor electrical penetrations are affected as a result of the increase of the brake horsepower to support unit operation at SPU conditions (see LR Section 2.2.2.6). The settings of the affected relays for the RCP hot and cold loop motor operation and reactor electrical penetration are being changed to address the increased bhp to support SPU conditions.

The evaluation confirms that there are no load changes to the existing Class 1E and common non-Class 1E 6.9 kV systems and remain bounded by the existing analyses for SPU conditions.

The evaluation of Class 1E and non-Class 1E switchgear busses, breakers, and associated feeder cables determined that they operate within their rated capability at SPU conditions. The evaluation also confirmed that the acceptance criteria, including limits for bus loading, voltage range, and short circuit current, are met at SPU conditions.

The evaluation of Class 1E and non-Class 1E 480V system determined that there are no 480V system load changes due to the SPU, and the 480V system loads are bound by those in the existing analyses. Therefore, the voltages, continuous currents, and fault currents for the substation transformers, busses, breakers, and feeder cables remain bounded for the SPU conditions.

The evaluation of the Class 1E and non-Class 1E low voltage AC busses and miscellaneous system loads for operation at the SPU conditions indicates that there are no load additions or modifications required to the low voltage AC system. Therefore, the evaluation concludes that there is no impact to the existing low voltage AC loading, voltage, short circuit feeder cables, and protection at SPU conditions.

The evaluation of the emergency diesel generator system and auxiliaries demonstrates that there are no changes to the emergency diesel generator loading or run time as a result of SPU conditions. The emergency diesel generator operation at SPU conditions remains bounded by the existing load analysis. The evaluation concludes that the emergency diesel generator system and it components have the capability and the capacity of performing their intended function at SPU conditions.

2.3.3.3 Conclusions

As a result of the evaluations for the SPU, Luminant Power has determined that the AC onsite power system steady-state and dynamic performance is acceptable with the implementation of the above described modifications. The electrical system load profile and the short circuit short levels remain acceptable for CPNPP. The AC onsite electrical equipment ratings capacities are sufficient to handle the proposed increase in generation capability at CPNPP. Adequate physical and electrical separation and equipment margin exists and the AC onsite power system has the capacity and capability to supply power to all safety loads and other required equipment at the SPU.

The evaluation concluded that the Units 1 and 2 AC onsite power system functions as designed and meets the current licensing basis with respect to the requirements of 10 CFR 50 Appendix A, GDC-17 following implementation of the proposed SPU.

2.3.4 DC Onsite Power System

2.3.4.1 Regulatory Evaluation

The plant direct current (DC) onsite power system is comprised of the DC power sources, distribution, and auxiliary supporting systems that supply motive or control power. The Comanche Peak Nuclear Power Plant (CPNPP) review covered the information, analyses, and referenced documents for the DC onsite power system. The Nuclear Regulatory Commission's (NRC's) acceptance criteria for the DC onsite power system are based on 10 CFR 50, Appendix A, General Design Criterion (GDC) -17. It requires the system to have the capacity and capability to perform its intended functions during anticipated operational occurrences and accident conditions.

Current Licensing Basis

Final Safety Analysis Report (FSAR) Section 8.3.2 states that the Class 1E DC power systems are designed with sufficient capacity, independence and redundancy to ensure performance of their safety functions assuming a single failure.

FSAR Sections 8.3.1.2.1 and 8.3.2.2 demonstrates that the Class 1E 125V DC power systems comply with the design requirements of 10 CFR 50, Appendix A, GDCs -17 and -18 and NRC Regulatory Guides 1.6, 1.32 and 1.75. The overall system design including functional requirements, redundancy, capacity, and availability is in conformance with Institute of Electrical and Electronics Engineers (IEEE) 308-1974 modified by the requirement that the battery performance discharge test intervals are in accordance with IEEE 450-1995.

2.3.4.2 Technical Evaluation

2.3.4.2.1 Introduction

The DC power system is discussed in FSAR Section 8.3.2. The DC systems for each unit consist of two independent and redundant Class 1E 125V battery systems and one 125V, one 125/250V, one 24/48V non-Class 1E battery systems. Each of 125V DC Class 1E power systems is completely independent of the other Class 1E power system. The Class 1E DC power systems provide power to Class 1E loads without interruption during normal operations or design basis accident (DBA) conditions. Non-Class 1E loads are supplied from the 125V DC, 125/250V DC, and 24/48V DC systems. All of the non-Class 1E systems are completely independent, both of the Class 1E 125V DC systems and of each other.

The Class 1E DC power systems provide power to Class 1E loads without interruption during normal operations or DBA conditions. Each unit Class 1E 125V DC power system consists of two trains, A and B. Each train consists of two independent busses. Each bus is supplied by one battery and two 100-percent capacity battery chargers (one spare). The battery and two battery chargers provide power to the corresponding switchboard. Two circuit breakers connecting these battery chargers to the DC distribution switchboard and are mechanically interlocked such that only one charger remains connected to the switchboard at any time. Each

battery charger is sized to recharge the battery from the design minimum charge state to the fully charged state within 24 hours while supplying the steady-state loads under all modes of plant operation. Input power to these chargers is obtained through independent 480V, three-phase, AC supply from Class 1E MCCs. Each Class 1E battery is capable of carrying the essential load continuously for a period of four hours in the event of a total loss of onsite and offsite AC power.

The Non-Class 1E 125V DC system is primarily a part of uninterruptible power supply (UPS) system used to supply the plant computer and peripherals. The system consists of a 125V battery, two battery chargers (one spare), and a main distribution bus (switchboard) with molded case circuit breakers. The 125V DC system supplies two inverters and a distribution panel.

The non-Class 1E 125/250V DC system consists of two 125V batteries, three 125V battery chargers (one spare), a main distribution bus (switchboard) with molded case circuit breakers, and fusible switches. The non-Class 1E 125/250V DC power system powers large DC loads such as turbine generator auxiliaries and control power for the non-Class 1E medium voltage switchgear.

The non-Class 1E 24/48V DC system for each unit consists of two 24V batteries, three battery chargers (one spare), main switchboard, distribution panels, molded case circuit breakers, and fusible switches. The 24/48V DC system powers turbine generator control and instrumentation.

Upon loss of AC power, non-safety related DC power systems provide power for emergency lighting. The DC emergency lighting system is described in FSAR Section 9.5.3.

Each Class 1E and non-Class 1E switchboard supplies power via molded case circuit breakers or fusible switches to associated distribution panels and inverters.

Each of the 125V DC Class 1E power systems is completely independent of the other Class 1E power systems. There are no bus ties or sharing of power supplies between redundant trains. Class 1E equipment associated with systems shared by both units receives power from panel boards having an incoming automatic transfer switch, which can select power from either unit.

Train separation is maintained by supplying these shared panel boards from the same train of both units. Independence and separation are maintained throughout the DC distribution network. The DC system provides the battery capacity to cope with station blackout and fire safe shutdown conditions.

2.3.4.2.2 Description of Analyses and Evaluations

The Class 1E and non-Class 1E DC power systems and their associated components were evaluated to ensure they are capable of performing the system's intended function at stretch power uprate (SPU) conditions. The evaluation is based on the system's required design functions and attributes, and upon a comparison between the existing DC equipment ratings and the anticipated operating requirements at SPU conditions.

Station Blackout and Fire Protection Program evaluations are included in Licensing Report (LR) subsections 2.3.5 and 2.5.1.4, respectively.

FSAR Sections 8.3.1.2.1 and 8.3.2.2 evaluate each DC power system and state that they comply with the intent of 10 CFR 50, Appendix A GDC-17 and -18, Reg. Guides 1.6, 1.32 and 1.75, IEEE 308-1974, and IEEE 450-1995.

2.3.4.2.3 Results

The Class 1E and non-Class 1E DC portions of the DC power systems were evaluated to determine potential impacts of the SPU.

No additional loads have been added to the DC systems and there have been no existing load changes as a result of SPU.

The capability and capacity of the DC system remain unchanged.

DC load changes, resulting from plant modifications, are evaluated as part of the design change process.

Station Blackout and Fire Protection Program evaluations did not result in any DC systems load changes as discussed in LR subsections 2.3.5 and 2.5.1.4.

2.3.4.3 Conclusions

The evaluation concluded that the Unit 1 and 2 Class 1E and non-Class 1E DC power systems will continue to function as designed and continue to meet the current licensing basis with respect to the requirements of 10 CFR 50 Appendix A, GDC 17 following implementation of the proposed SPU. Adequate physical and electrical separations exist, and the DC system has the capacity and capability to supply power to all safety loads and other required equipment at SPU conditions.

2.3.5 Station Blackout

2.3.5.1 Regulatory Evaluation

Station blackout (SBO) refers to a complete loss of alternating current (AC) electric power to the essential and nonessential switchgear busses in a nuclear power plant. Station blackout involves the loss of offsite power concurrent with a turbine trip and failure of the onsite emergency AC power system. Station blackout does not include the loss of available AC power to busses fed by station batteries through inverters or the loss of power from "alternate AC sources." This review focused on the impact of the proposed stretch power uprate (SPU) on the plant's ability to cope with and recover from a station blackout event for the period of time established in the plant's licensing basis.

Current Licensing Basis

The adequacy of the Comanche Peak Nuclear Power Plant (CPNPP) design relative to the General Design Criteria (GDC) is discussed in Final Safety Analysis Report (FSAR) Sections 3.1.1 and 3.1.2. CPNPP was evaluated against the requirements of the Station Blackout Rule, 10 CFR 50.63 using Regulatory Guide 1.155; the blackout coping duration was determined to be 4 hours.

As addressed in FSAR Appendix 8B, station blackout assumptions as presented in NUMARC 87-00 stipulate that following the loss of offsite power the reactor is assumed to automatically trip with sufficient shutdown margin to maintain subcriticality at Mode 3 (Hot Standby) or Mode 4 (Hot Shutdown).

An emergency diesel generator on the non-SBO unit will energize on the loss-of-offsite power and provide power to common equipment in support of the SBO unit. However, as this is a normal electrical line up, it does not constitute the "alternate AC" approach to coping with SBO.

2.3.5.2 Technical Evaluation

The station blackout rule requires the following issues be addressed: station blackout duration, condensate inventory for decay heat removal, Class 1E battery capacity, compressed air, effects of loss of ventilation, containment isolation, reactor coolant inventory, procedures and training, quality assurance and Technical Specifications, and the emergency diesel generator reliability program.

2.3.5.2.1 Introduction

A summary of the status of the following SBO issues for current plant conditions is provided, since these are potentially affected by the SPU: condensate inventory for decay heat removal, Class 1E battery capacity, compressed air, effects of loss of ventilation, containment isolation, reactor coolant inventory, plant procedures and training, and auxiliary feedwater system flow requirements for SBO.

Condensate Inventory for Decay Heat Removal

The required condensate inventory for SBO will increase, however, the inventory at the uprate licensed power level for decay heat removal and plant cooldown required by Technical Specification 3.7.6 (see Licensing Report (LR) subsection 2.5.4.5) bounds the SBO required volume.

Class 1E Battery Capacity

Each Class 1E battery and its distribution system is designed and installed to provide a reliable source of redundant onsite direct current (DC) power. Each Class 1E battery is capable of supplying power for four hours to the Class 1E loads connected to the Class 1E bus which it serves.

Compressed Air

The air-operated valves and dampers fail in a safe position on loss of air. Air-operated valves that are needed for SBO can be initially operated remotely with air stored in accumulators and locally operated manually when air is exhausted.

Effects of Loss of Ventilation

The uninterruptible power supply (UPS) and distribution rooms, the turbine-driven auxiliary feedwater pump area, the battery rooms, and the switchgear rooms were also identified as containing SBO equipment. The following conservative maximum temperatures for a 4-hour coping period were determined:

- UPS and distribution rooms: < 122°F
- Turbine-driven auxiliary feedwater pump area: 131.1°F
- Minimum temperature for the battery rooms on loss of ventilation: 67°F
- Switchgear rooms: 104°F

Containment Isolation

An assessment was performed confirming that containment integrity can be provided during an SBO event, where "containment integrity" is defined as providing the capability for valve position indication and closure of certain containment isolation valves independent of the preferred or Class 1E power supplies.

Plant Procedures and Training

An emergency operating procedure, "Loss of All AC Power," provides the sequence and instructions for operating SBO coping equipment.

Auxiliary Feedwater Flow

An analysis performed to determine the auxiliary feedwater flow rate for current plant conditions required to remove decay heat and cooldown following a station blackout event showed that a flow rate to each steam generator from the turbine-driven auxiliary feedwater pump was sufficient.

2.3.5.2.2 Acceptance Criteria

The Station Blackout Rule, 10 CFR 50.63, identifies the factors that must be considered in specifying the station blackout duration. It requires that, for the station blackout duration, the plant be capable of maintaining core cooling and appropriate containment integrity. It also addresses requirements for station batteries, and requirements for alternate AC power sources.

2.3.5.2.3 Description of Analyses and Evaluations

Condensate Inventory for Decay Heat Removal

The required condensate inventory at the uprate licensed power level for decay heat removal and plant cooldown is bounded by the condensate inventory in Technical Specification 3.7.6.

Class 1E Battery Capacity

Each Class 1E station battery for Unit 1 and 2 is designed to carry the connected loads continuously for a period of four hours in the event of a loss of onsite or offsite AC power. Accordingly, the station vital batteries are not affected by the SPU and continue to have sufficient capacity to meet station blackout loads for the 4 hour coping duration at SPU conditions.

Compressed Air

The air-operated valves required by SBO to have accumulators to provide backup air supply upon loss of instrument air are the auxiliary feedwater flow control valves and the steam generator atmospheric relief valves. The accumulators at each auxiliary feedwater control valve are sized on the basis of allowing the operator remote manual control to isolate a faulted steam generator for a period of 30 minutes after loss of air. Flow control is performed locally after the accumulator air is exhausted. Each steam generator atmospheric relief valve accumulator can provide for 15 positionings over a 4-hour period after loss of instrument air. All valves are accessible for local operation during SBO.

Effects of Loss of Ventilation

A calculation of the temperature response of a large dry containment like CPNPP indicates that temperatures inside containment, resulting from the loss of ventilation during an SBO, are enveloped by the loss-of-coolant accident and main steam line break temperature profiles.

Similarly, for equipment outside of containment, areas containing equipment required to cope with an SBO need to be evaluated only if the area is a dominant area of concern and the dominant area of concern has not been previously evaluated as a harsh environment due to a high-or moderate-energy line break. However, at CPNPP, equipment required to cope with SBO has been evaluated for operability. There is no impact due to the SPU.

The operator actions specified in the "Loss of All AC Power" procedure continue to be applicable for SPU conditions.

Containment Isolation

The SPU does not add or remove any containment isolation valves. The ability to close required valves and the required valve position indication capability is not related to power level or other SPU-related changes. Accordingly, the evaluation of this issue for current plant conditions remains applicable for SPU conditions.

Plant Procedures and Training

The SBO event emergency operating procedure, "Loss of All AC Power," and the procedure addressing alternate water supply to the auxiliary feedwater pumps are not affected by the SPU. The operator actions specified in the "Loss of All AC Power" procedure continue to be applicable for SPU conditions.

Auxiliary Feedwater Flow

The only parameter in the auxiliary feedwater flow rate analysis for current plant conditions affected by the SPU is the decay heat associated with the uprated core power level (increase from 3,458 to 3,612 MWt). A flow rate sufficient for SPU conditions is within the capacity of the turbine-driven auxiliary feedwater pump (LR subsection 2.5.4.5, Auxiliary Feedwater System).

2.3.5.3 Conclusion

The effects of the proposed SPU on the plant's ability to cope with and recover from an SBO event for the period of time established in the plant's licensing basis have been reviewed. It is concluded that the effects of the proposed SPU on SBO have been adequately evaluated and it has been demonstrated that the plant will continue to meet the requirements of 10 CFR 50.63 following implementation of the proposed SPU. Therefore, the proposed SPU is acceptable with respect to SBO.

2.4 INSTRUMENTATION AND CONTROL

2.4.1 Reactor Protection, Engineered Safety Feature Actuation, and Control Systems

2.4.1.1 Regulatory Evaluation

Instrumentation and control (I&C) systems are provided (1) to control plant processes having a significant impact on plant safety, (2) to initiate the reactivity control system (including control rods), (3) to initiate the engineered safety features (ESF) systems and essential auxiliary supporting systems, and (4) for use to achieve and maintain a safe shutdown condition of the plant. Diverse instrumentation and control systems and equipment are provided for the express purpose of protecting against potential common-mode failures of instrumentation and control protection systems.

Luminant Power reviewed the following systems for the proposed stretch power uprate (SPU) to ensure that the systems and any changes necessary for the proposed SPU are adequately designed such that the systems continue to meet their safety functions:

- Reactor trip system (RTS)
- ESF actuation system (ESFAS)
- Safe shutdown systems
- Control systems

The review was also conducted to ensure that failures of these systems do not affect safety functions.

The acceptance criteria are based on 10 CFR Part 50.55a (a)(1); 10 CFR Part 50.55a(h); and:

- General Design Criterion (GDC) -1, insofar as it requires that structures, systems, and components (SSCs) important to safety be designed, fabricated, erected, constructed, and tested to quality standards commensurate with their importance to performed functions.

- GDC-4, insofar as it requires that SSCs be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.

- GDC-13, insofar as it requires that instrumentation is provided to monitor variables and systems over their anticipated ranges for normal operation, anticipated operational occurrences, and for accident conditions as appropriate to ensure safety, including those variables and systems that can affect the fission process, reactor core integrity, the reactor coolant pressure boundary (RCPB), and the containment and its associated systems. Appropriate controls should be provided to maintain these variables and systems within prescribed operating ranges.

- GDC-19, insofar as it requires that a Control Room be provided from which actions can be taken to safely operate the nuclear unit under normal conditions, and maintain it in a safe condition under accident conditions, including loss-of-coolant accidents (LOCAs).

- GDC-20, insofar as it requires that the protection system be designed to (1) automatically initiate the operation of appropriate subsystems, including the reactivity control systems, to ensure that acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences, and (2) to sense accident conditions and automatically initiate operation of systems and components important to safety.

- GDC-21, insofar as it requires that protection systems be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function, and (2) removal from service if any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated.

- GDC-22, insofar as it requires that the protection systems be designed to ensure that the effects of natural phenomena and normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis.

- GDC-23, insofar as it requires that protection systems be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as system disconnection, loss of energy (such as electric power or instrument air), or postulated adverse environments (such as extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

- GDC-24, insofar as it requires that the protection systems be separated from the control systems to the extent that a system satisfying all reliability, redundancy, and independence requirements of the protection systems is left intact in the event of a failure of any single control system component or channel, or failure or removal from service of any single control system component or channel that is common to the control and protection systems. Protection and control system interconnection will be limited to ensure that safety is not significantly impaired.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the CPNPP reactor protection, engineered safety feature actuation, and control systems adequacy regarding conformance to:

- GDC-1, Quality Standards and Records, is described in FSAR Section 3.1.1.1.

  FSAR Chapter 17 provides direct reference to the Quality Assurance (QA) Program established to provide assurance that safety related SSCs satisfactorily perform their intended safety functions. The QA Program conforms to the intent of 10 CFR Part 50, Appendix B, Quality Assurance Criteria for Nuclear Power plants and Fuel Reprocessing Plants. Those items for which the requirements of 10 CFR Part 50, Appendix B are met are listed in the list of quality assured items in FSAR Appendix 17A.

- GDC-4, Environmental and Missile Design Bases, is described in FSAR Section 3.1.1.4.

  Physical separation, physical protection, pipe restraints, and redundancy are included in the design of safety-related systems to ensure that each such system performs its intended safety function.

  The stations SSCs important to safety are designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents including LOCAs. Environmental conditions are described in FSAR Section 3.11.

- GDC-13, Instrumentation and Control, is described in FSAR Section 3.1.2.4.

  To ensure adequate safety, instrumentation and controls are provided to monitor and control significant variables over their anticipated range for all conditions in the reactor coolant system (RCS), steam and power conversion system, radioactive waste systems, ESFs, and the Containment Building. Parameters that must be provided for operator use under normal operating and accident conditions are indicated in the Control Room in close proximity to the controls which maintain the indicated parameters in the proper range.

  The quantity and types of processing instrumentation provided ensures safe and orderly operation of all systems over the full design range of the plant. These systems are described in FSAR Chapters 6, 7, 8, 9, 11, and 12.

- GDC-19, Control Room, is described in FSAR Section 3.1.2.10.

  Safe occupancy of the Control Room during abnormal conditions is provided for in the design.

  The Control Room is continuously occupied by qualified operating personnel under all operating and accident conditions.

In the unlikely event that access to the Control Room is restricted, the hot shutdown panel, local control stations, or manual operation of critical components can be used to effect hot shutdown from outside the Control Room for an extended period.

Before evacuation takes place, the reactor can be manually tripped and neutron level and control rod position can be verified. By use of appropriate procedures and equipment, the unit can also be brought to cold shutdown conditions (see FSAR Section 7.4).

- GDC-20, Protection System Functions, is described in FSAR Section 3.1.3.1.

  A fully automatic protection system, with appropriate redundant channels, is provided to cope with transients where insufficient time is available for manual corrective action. The design basis for all protection systems is in accordance with the intent of Institute of Electrical and Electronic Engineers (IEEE) 279-1971 and IEEE Standard 379-1972. The reactor protection system automatically initiates a reactor trip when any variable monitored by the system or combination of monitored variables exceeds the normal operating range. Setpoints are designed to provide an envelope of safe operating conditions with adequate margin for uncertainties to ensure that fuel design limits are not exceeded.

  Reactor trip is initiated by removing power to the control rod drive (CRDMs) mechanisms of all full-length rod cluster control assemblies (RCCAs). This causes the rods to insert by gravity rapidly reducing reactor power output. The response and adequacy of the protection system have been verified by analysis of anticipated transients.

  Refer to FSAR Chapter 7, Instrumentation and Controls, for additional information regarding actuating devices to the protection system.

  The ESFAS automatically initiates emergency core cooling and other safeguards functions by sensing accident conditions using redundant analog channels measuring diverse variables. In addition, manual action of safeguards equipment can be performed where ample time is available for operator action. In either case, the ESFAS automatically trips the reactor on manual or automatic safety injection signal generation.

- GDC-21, Protection System Reliability and Testability, is described in FSAR Section 3.1.3.2.

  The protection system is designed for high functional reliability and in-service testability. It is designed with sufficient redundancy and independence to permit acceptable reliability of operation. System design also includes the capability to perform periodic testing of channels independently while the reactor is in operation.

  Compliance with this criterion is discussed in detail in FSAR Sections 7.2.2.2.3 and 7.3.2.2.5.

- GDC-22, Protection System Independence, is described in FSAR Section 3.1.3.3.

  Protection system components are designed and arranged so that there is no loss of safety function in the environment accompanying any emergency situation in which the components are required to function. Various means are used to accomplish this. Functional diversity has been designed into the system. The extent of this functional diversity has been evaluated for a wide variety of postulated accidents. Also, diverse protection functions will automatically terminate an accident before intolerable consequences can occur.

  Automatic reactor trips are based on neutron flux measurements, reactor coolant loop temperature measurements, pressurizer pressure and level measurements, and reactor coolant pump power underfrequency and undervoltage measurements. Trips can also be initiated manually by safety injection signal (see FSAR Section 7.2).

  High-quality components, conservative design, and applicable quality control, inspection, calibration, and tests are used to guard against common-mode failure. Qualification testing is performed on the various safety systems to demonstrate functional operation at normal and post-accident conditions of temperature, humidity, pressure, chemistry, and radiation for specific periods, if required. See FSAR Section 3.11 for further details. Typical protection system equipment is subjected to tests under simulated seismic conditions using conservatively large accelerations and applicable frequencies. The test results indicate no loss of the protection function. See FSAR Section 3.10 for further details.

- GDC-23, Protection System Failure Modes, is described in FSAR Section 3.1.3.4.

  The protection system is designed with due consideration of the most probable failure modes of the components under various perturbations of the environment and energy sources. Each reactor trip channel is designed on the de-energize-to-trip principle so that loss of power disconnection, open channel faults, and the majority of internal channel-short-circuit faults cause the channel to go into its tripped mode.

  The protection system is discussed in FSAR Sections 7.2 and 7.3.

- GDC-24, Separation of Protection and Control Systems, is described in FSAR Section 3.1.3.5.

  The protection system is separate and distinct from the control systems. Control systems may be dependent on the protection system because control signals are derived from protection system measurements, where applicable. These signals are transferred to the control system by isolation amplifiers, which are classified as protection components. The adequacy of system isolation has been verified by testing under conditions of postulated credible faults. The failure of a single control system component or channel, or failure or removal from service of any single protection system component or channel, that is common to the control and protection systems, leaves

intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. See FSAR Chapter 7 for details. Distinction between channel and train is made in this discussion. The removal of a train from service is allowed only during testing of the train.

FSAR Chapter 7 discusses I&C systems. The primary purpose of the I&C systems is to provide automatic protection and exercise proper control against unsafe and improper reactor operations during steady-state and transient power operations (American Nuclear Society (ANS) Conditions I, II, and III), and to provide initiating signals to mitigate the consequences of faulted conditions (ANS Condition IV). ANS conditions are discussed in FSAR Chapter 15. The information presented in FSAR Chapter 7 emphasizes those I&C systems that are central to ensuring that the reactor can be operated to produce power in a manner that ensures no undue risk to the health and safety of the public.

Other FSAR sections that address the design features and functions of plant safety-related systems and instrumentation include:

- FSAR Section 7.1.1.1, Safety Related Systems, describes the instrumentation that is required to function to achieve the system responses assumed in safety evaluations, and those instruments needed to safely shut down the plant.

- FSAR Section 7.1.2, Identification of Safety Criteria, provides the design bases in FSAR Section 7.1.2.1 for the systems listed in FSAR Section 7.1.1.1.

- FSAR Section 7.2, Reactor and Trip System, provides the system description (including functional performance requirements, reactor trips, interlocks and setpoints), design bases, and analyses (including control and protection system interaction).

- FSAR Section 7.3, Engineered Safety Features Systems, provides the system description, design bases (including limits, margins and setpoints), and analyses (including control and protection system interaction).

- FSAR Section 7.4, Systems Required for Safe Shutdown, identifies the minimum systems required to achieve and maintain cold shutdown without offsite power, and with an event initiated by a single random failure.

- FSAR Section 7.5, Information Systems Important to Safety, identifies the instrumentation available to the operator in the event that the integrity of the in-core fuel cladding, RCS pressure boundary, or the reactor containment has degraded beyond the prescribed limits defined as in the plant safety analyses.

- FSAR Section 7.6, All Other Instrumentation Systems Required for Safety, describes the instrumentation and control power supply system, residual heat removal isolation valves, refueling interlocks, accumulator motor-operated valves, switchover from injection to recirculation, process and effluent radiological monitors, RCPB leakage detection

systems, interlocks for RCS pressure control during low-temperature operation, monitoring combustible gas in the containment, fire detection system, instrumentation for mitigating consequences of inadvertent boron dilution, and mitigation of environmental effects of pipe breaks outside containment.

- FSAR Section 7.7, Control Systems Not Required for Safety, provides a description of the reactor control system, rod control, plant control signals for monitoring and indicating, plant control system interlocks, pressurizer pressure and water level control, steam generator water level control, steam dump, and in-core instrumentation. Also included is a description of the plant response to design loading and unloading.

- FSAR Section 7.8, Anticipated Transient Without Scram Mitigation System Actuation Circuitry (AMSAC), provides a description of the AMSAC system that provides a backup to the reactor protection system and ESFAS for initiating turbine trip and auxiliary feed flow in the event of an anticipated transient (such as complete loss of main feed water).

2.4.1.2 Technical Evaluation

2.4.1.2.1 Introduction

The CPNPP safety analyses are being converted to Westinghouse methodologies as described in License Amendment Request (LAR) 07-003 submitted in letter TXX-07063 and the cycle specific changes as provided in LAR 07-006 submitted in letter TXX-07108. These safety analyses were performed at SPU conditions. The SPU is based on approval of the license changes for Westinghouse methodology including the cycle specific changes provided in the above identified letters. Therefore, the below discussion is a summary of the impact of the above revised analyses as they relate to the reactor trip system, ESFAS, and the reactor control system.

With respect to the SPU, the reactor trip system, ESFAS, and the reactor control systems, may be impacted by the increase in reactor thermal power from 3,458 to 3,612 MWt. Unit 1 and Unit 2 were each assessed and differences are noted.

2.4.1.2.2 Input Parameters and Assumptions

The design parameters associated with the uprate are identified in Licensing Report (LR) Section 1.1, Nuclear Steam Supply System Parameters.

2.4.1.2.3 Analyses and Evaluations

The effects of the increase in reactor thermal power have been evaluated for normal operation, operational transients, and accident conditions described in FSAR Sections 6.0, Engineered Safety Systems; 7.7.1.1, Reactor Control System; and 15, Accidents Analyses. These analyses used the most conservative combination of nuclear steam supply system (NSSS) design values from LR Section 1.1, Nuclear Steam Supply Steam Parameters. The results of the transient and accident analyses are described in the following LR sections:

- 2.4.2.1, Plant Operability
- 2.6, Containment Review Considerations
- 2.8.5, Accident and Transient Analyses

The safety analyses identified trip setpoint changes that are required to ensure departure from nucleate boiling (DNB), RCS pressure, and secondary system pressure remains within the allowable design margins and the response to the design basis operational transients remains acceptable. These changes are further described in this Licensing Report section.

The SPU analyses determined that, the NSSS instrumentation ranges, scaling, and setpoints used in the reactor protection, ESFAS, and reactor control instrumentation remained adequate for the SPU.

Description of Balance-of-Plant Analyses and Evaluation

Operation of the plant at SPU has minimal effect on balance-of-plant (BOP) system instrumentation and control devices. Based on SPU operating conditions for the power conversion and auxiliary systems, process control valves and instrumentation have sufficient range/adjustment capability for use at the SPU conditions except as noted below.

The evaluation methodology used to evaluate the BOP system instrumentation includes the following basic steps:

- Perform system analysis to determine how the SPU conditions/ranges/setpoints compare to the current operating conditions/ranges/setpoints for the BOP systems.

- For those systems (subsystems) that are impacted by the SPU, determine the major process instrumentation or board-mounted instruments from the piping and instrument diagrams (P&IDs) and instrument scaling calculations and tabulate the pre-SPU and post-SPU process data.

- Analyze the affected instruments to determine SPU instrument impact.

- For those instruments affected by the SPU, recommend new scaling, setpoints, ranges, or a suitable replacement (if required).

Balance-of-Plant Instrumentation and Controls

BOP instrumentation evaluated (except items that are part of or provide input to the NSSS and/or the main turbine control system) included the following fluid systems:

- Condensate system (FSAR Section 10.4.7)
- Heater drain system (FSAR Section 10.4.11)
- Circulating water system (FSAR Section 10.4.5)
- Turbine plant cooling water system (FSAR Section 10.4.12)
- Steam dump system (FSAR Section 10.4.4)
- Turbine generator system (FSAR Section 10.2)
- Extraction steam system (FSAR Section 10.4.10)
- Feedwater system (upstream of isolation valves) (FSAR Section 10.4.7)
- Feedwater system blowdown system (FSAR Section 10.4.8)
- Spent fuel pool cooling and cleanup system (FSAR Section 9.3.1)

The SPU evaluation of BOP instrumentation and controls demonstrated that, except as noted below, the design of BOP instruments, ranges, and setpoints remains acceptable for SPU operation.

The existing BOP indicated spans for indicators located at the hot shutdown panel for monitoring steam line pressure and condensate storage tank level, as identified in FSAR Section 7.4.2.1, are unaffected at SPU conditions.

The Regulatory Guide (RG) 1.97, Rev. 2 monitored variables, as identified in FSAR Section 7.5 and summarized in Table 7.5-7A, 7.5-7B, and Table 7.5-7D, which identify specific deviations from the guidance in RG 1.97, Rev. 2, for monitoring BOP variables remain bounding at SPU conditions. These variables are as follows:

- Main steam line pressure
- Main feedwater flow
- Condensate storage tank level
- Component cooling water (CCW) header temperature
- Service water header flow
- CCW header pressure
- CCW surge tank level

The flow indicating switches and related setpoints used for the detection of a steam generator blowdown (SGB) line break in any of the four SGB Lines, as identified in FSAR Section 7.6.12, are unaffected at SPU conditions.

- Condensate and Feedwater System

  The condensate and feedwater system evaluation is described in LR subsection 2.5.5.4, Feedwater and Condensate. As a result of this evaluation, the following modifications will be implemented:

  - Change the existing scaling for the main feed pump turbine speed control loops.

- To maintain operating margin at SPU conditions, the following setpoints and Alarm Response Procedures will be changed:
  - Feedwater pump low feedwater pump suction pressure, open condensate polishing bypass valve.
  - Feedwater pump low feedwater pump suction pressure, open low pressure heater bypass valve.
  - Feedwater pump low feedwater pump suction pressure, trip main feedwater pump.
- Change the Control Room indicator banding for condensate flow to the gland steam condenser.
- Change the Control Room indicator banding and rescale the instrument loops for main feedwater flow.
- Change the Control Room indicator banding and recalibrate/rescale the instrument loop for main feedwater pump suction flow.
- Change the Control Room indicator banding for blowdown heat exchanger condensate water outlet temperature.
- Change the Control Room indicator banding and recalibrate/rescale the instrument loop for heater drain pump discharge flow.

- Main Steam System

  The main steam system evaluation is described in LR subsection 2.5.5.1, Main Steam. As a result of this evaluation, the following modifications will be implemented.

  - Change the Control Room indicator banding for main steam flow.
  - Recalibrate/rescale the instrument loops for moisture separator reheater steam supply flow.
  - Recalibrate/rescale the instrument loops for turbine first stage pressure.
    - Instruments associated with turbine first stage pressure will require scaling changes for various nuclear steam supply system (NSSS) inputs (anticipated transient without scram (ATWS) mitigation system actuation circuitry (AMSAC)), Permissive P13, rod control, block auto rod withdrawal, and steam dump control)

- Extraction Steam System

  The extraction steam system evaluation is described in LR subsection 2.5.1.2.2, Turbine Generator. As a result of this evaluation, the following modifications will be implemented.

  Change the Control Room indicator banding for both first and second point heater extraction steam pressure.

BOP Instrumentation and Controls Results

The changes to ranges and/or setpoints for BOP instruments will not change instrument or instrument loop functions. As a result of the SPU, there are no changes to the GDC-13 current licensing basis that the quantity and types of process instrumentation provided ensures safe and orderly operation of the plant nor will the changes affect separation, redundancy, or diversity of the instrumentation and controls discussed above.

Plant Computer

The plant computer (also referred to as the plant process computer system) is described in FSAR Section 7.7.1.11. Plant process computer system inputs that are affected by instrumentation scaling changes will be modified during the implementation phase of the SPU. However, the plant computer functions as described in FSAR Section 7.7.1.11 and will not change as result of the SPU.

Nuclear Steam Supply System Instrument and Controls

Technical Specification limiting safety system setting (LSSS) values and trip setpoint values are derived from analytical values used in analyses corrected to account for the specific instrument or control system uncertainty.

Sensor and rack error terms for calibration accuracy and drift are grouped in the channel statistical allowance equation with their dependent measurement and test equipment (M&TE) terms, then combined with the other independent error terms using the square-root-sum-of-the-squares (SRSS) methodology.

The design bases and description of the CPNPP RTS is described in FSAR Section 7.2, Reactor and Trip System (RTS), and includes a listing of the reactor trips, purpose of each trip, and any associated protection and control permissives. The RTS automatically trips the reactor to protect against RCS damage caused by high system pressure and to protect the reactor core against fuel rod cladding damage caused by a DNB. The basic reactor tripping philosophy is to define a region of power and coolant temperature and pressure conditions allowed by the primary trip functions (OP N-16 trip, nuclear overpower OT N-16 trip, and pressurizer pressure high and low trips). The allowable operating region within these trip settings is provided to prevent any combination of power, temperature, and pressure that would result in a DNB with all reactor coolant pumps in operation.

Additional trip functions such as a high pressurizer water level trip, loss-of-flow trip, steam generator low-low water level trip, turbine trip, safety injection trip, nuclear source and intermediate range trips, and manual trip are provided to back up the primary trip functions for specific accident conditions and mechanical failures.

The following is a list of the RTS instrumentation and setpoint impacts necessary to ensure the RTS will continue to satisfy its design functions at SPU conditions.

Nuclear Instrumentation

SPU redefines the 100-percent power neutron flux levels and will impact the flux level to percent power relationship for the intermediate range and power range nuclear instruments. Since the source range nuclear instrumentation is deenergized well below the power range, during reactor startup, there are no changes required to the source range instrumentation settings. The SPU accident and transient analyses determined that for all accidents that explicitly credit power range high flux trip setpoint, the analytical limit for the power range high power trip could be increased from the current setpoint. With the power range high power trip safety analysis limit (SAL), the current field trip setpoint of 109 percent has adequate margin to accommodate the new SAL limit and will not change.

The accident and transient analyses also determined the analytical limit for the power range low power reactor trip at ≤ 34.5 percent of rated thermal power (RTP) can be increased for SPU to 35 percent. Therefore, the current power range low power reactor trip setpoint (25 percent) remains adequate for SPU. In addition, the intermediate range reactor trip setpoint (25 percent) will remain adequate for SPU.

RCS Temperature Instrumentation

No changes are recommended for the T_{hot}, T_{cold}, T_{avg}, and N-16 instrument ranges for the SPU conditions. As such, the existing values are adequate to provide the required indication, core DNB protection, and plant response during accidents and transients over the entire range of operation at SPU conditions.

Overtemperature N-16 Trip

Typically, the values for the overtemperature N-16 trip setpoints constants are listed in the cycle-specific Core Operating Limits Report (COLR) for each fuel cycle.

Overpower N-16 Trip

The Nominal Trip Setpoint and Allowable Value for the overpower N-16 reactor trip function is being revised as described in License Amendment Request 07-006, submitted in letter TXX-07108, which proposes cycle-specific Technical Specification changes required to support the transition of the accident analysis methodologies to standard Westinghouse methodologies. The SPU accident and transient analyses determined that the proposed setpoints are valid for the SPU conditions.

Steam Generator Water Level Instrumentation

The accident and transient analyses have determined that the analytical limits utilized in the loss of normal feedwater event and the feedline break analyses are increasing for the SPU. The steam generator water level low-low trip (and ESFAS initiation) safety analyses limit is increasing from 0-percent narrow-range span (NRS) to 10-percent NRS for a loss of normal feedwater for Unit 2 only. Unit 1 remains at 0-percent NRS. For the feedline break analyses, the safety analysis limit for Unit 1 is increasing from 0-percent NRS to 10-percent NRS, and for Unit 2 is increasing from 0-percent NRS to 7.5-percent NRS. The field setpoint has adequate margin to accommodate the new SAL limit and will not change.

Anticipated-Transient-Without-Scram Mitigation System Actuation Circuitry (AMSAC)

The AMSAC, as required by 10 CFR 50.62, is described in FSAR Section 7.8, Anticipated-Transient- Without-Scram Mitigation System Actuation Circuitry. The SPU (LR subsection 2.8.5.7) shows that the current setpoints do not require any change due to SPU. Normal scaling will be performed per existing plant procedures.

Permissive Changes

Permissives requiring input from the turbine first-stage pressure will be recalibrated, if necessary, to be consistent with the turbine first-stage pressure range.

2.4.1.2.3.1 Engineered Safety Feature Actuation System

The ESFAS is used to provide protection against the release of radioactive materials in the event of a LOCA or a secondary line break accident. The ESF system functions to maintain the reactor in a shutdown condition. It also provides sufficient core cooling to limit the extent of fuel and fuel cladding damage and to ensure the integrity of the containment structure. These functions rely on the ESFAS and associated instrumentation and controls. The following identifies the changes to the ESFAS instrumentation, analytical limits, and settings being implemented as part of SPU.

Additional Changes to ESFAS Analytical Limits

The following is the specific change to the ESFAS analytical values used in the accident analyses to support the SPU.

Steam Line Low Safety Injection

The analysis lead/lag time constant on the low steam line safety injection (SI) signal is 10/5 for the SPU.

2.4.1.2.3.2 Control Systems

The various reactor control systems are described in FSAR Section 7.7.1, Control Systems Not Required For Safety. The reactor control systems are designed to limit nuclear plant transients for prescribed design load perturbations, under automatic control, within prescribed limits to preclude the possibility of a reactor trip in the course of these transients. During steady-state operation, the primary function of the reactor control is to maintain a programmed average reactor coolant temperature that rises in proportion to load. The control systems also limit nuclear plant system transients to prescribed limits about this programmed temperature for specified load perturbations. Complete supervision of both the nuclear and turbine generator plants is accomplished from the Control Room. This supervision includes the capability to test periodically the operability of the reactor protection system.

The analyses evaluating the response to design basis operational transients at SPU conditions are described in LR subsection 2.4.2.1, Plant Operability. The acceptable response to the design basis operation transients and accidents and transients associated control system failures are based on the changes described for the rod control system and steam dump system being implemented.

Rod Control System Changes

The CPNPP rod control system is described in FSAR Section 7.7.1.2.

The rod control system responds to changes in RCS temperature and secondary load as sensed by the RCS measured T_{avg} instrumentation and turbine first-stage pressure instrumentation. The following changes were made to the rod control system setpoints for the CPNPP Units 1 and 2 SPU Program:

- The coolant average temperature program must be modified each time the full-load vessel average temperature is modified. This program is linear between no load and full load and is specified by setting the vessel average temperature (low limit) at no-load and full-load temperatures (high limit). For CPNPP Units 1 and 2, the vessel average temperature at no load is 557°F and the vessel average temperature at full load is between 574.2° and 589.2°F for a given fuel cycle. Table 2.4.1.2-5 provides the coolant average temperature program for two full-load temperatures of 574.2° and 589.2°F. For other full-power coolant average temperatures, the coolant average temperature program should be determined as follows:

  - High limit: Equivalent 100-percent power T_{avg} for a particular cycle
  - Low limit: 557°F
  - Full-power temperature: Equivalent 100-percent power T_{avg} for a particular cycle
  - Hot standby/no load: 557°F
  - Temperature gain: (Full Power Temperature, °F – 557°F)/100-percent power

- The lag time constant on the coolant average temperature program is 10 seconds.

- The lead/lag time constant on the coolant average temperature is 40/10 seconds.
- The power mismatch channel nonlinear gain breakpoint is 2 percent.

Pressurizer Water Level Program

The CPNPP Pressurizer Water Level Control Program is described in FSAR Section 7.7.1.5. The pressurizer water level control system maintains the pressurizer level within a programmed band consistent with measured average T_{avg}. The programmed level is designed to maintain sufficient margin above the low-level heater-cutoff setpoint and to maintain sufficient steam volume to ensure the pressurizer does not go solid during accidents and transient conditions.

Analyses described in LR subsection 2.4.2.1, Plant Operability, and LR subsection 2.8.5, Accident and Transient Analyses, are based the current nominal pressurizer level program of 25 – 60-percent for full-load T_{avg} values greater than or equal to 584.7°F and the following revised pressurizer level program for full-load T_{avg} values less than 584.7°F. The level program is linear between the low limit at the no-load temperature and the high limit at the full-load temperature. The current instrumentation is adequate to support operation at the uprated power.

- Pressurizer level program for full-load $T_{avg} \geq 574.2$°F and < 584.7°F

  Low limit = 25.0 percent of span
  High limit = ((1.581 percent/°F) * (full-load T_{avg}) – 864.4) percent of span

Steam Dump Control and Turbine Bypass Systems

The CPNPP steam dump control system is described in FSAR Section 7.7.1.7.

The evaluation of the steam bypass system is described in LR subsection 2.5.5.3, Turbine Bypass, and LR subsection 2.4.2.1, Plant Operability.

As described in LR subsection 2.4.2.1, Plant Operability, the current steam dump valve capacity is sufficient to accommodate a large load rejection equivalent to 50-percent RTP or a turbine trip at less than 50-percent RTP at SPU conditions for full-load T_{avg} values greater than or equal to 580°F with the current steam dump control system loss of load controller setpoints and for reduced full-load T_{avg} values less than 580°F with revised loss of load controller setpoints. Table 2.4.1.2-1 summarizes the current and revised loss of load controller setpoints for these two ranges of full load T_{avg}.

Furthermore, a reactor trip transient can be accommodated at SPU conditions for the full-load T_{avg} window (574.2° to 589.2°F) provided that the steam dump control system setpoints changes shown in LR Table 2.4.1.2-2 are implemented in the steam dump control system (plant trip controller).

Steam Generator Water Level Control

The steam generator level control system is described in FSAR Section 7.7.1.7, Steam Generator Water Level Control (SGWLC). Each steam generator is equipped with a three-element feedwater flow controller that maintains a programmed water level that is a function of turbine load. The three-element feedwater controller regulates the feedwater valve by continuously comparing the feedwater flow signal, the water level signal, the programmed level and the pressure compensated steam flow signal. In addition, the main turbine feedwater pump speed is varied to maintain a programmed pressure differential between the steam header and the feed pump discharge header. The speed controller continuously compares the actual ΔP with a programmed ΔP_{ref} which is a linear function of steam flow. Continued delivery of feedwater to the steam generators is required as a sink for the heat stored and generated in the reactor following a reactor trip and turbine trip. An override signal closes the feedwater valves when the average coolant temperature is below a given temperature and the reactor has tripped. Manual override of the feedwater control system is available at all times. The SGWLC is acceptable for operation at SPU conditions.

In-Core Instrumentation

The in-core thermal thermocouples (TCs) and in-core movable detectors are described in FSAR Section 7.7.1. With respect to SPU, the in-core TCs will be exposed to higher core exit temperatures. However, these temperatures are well within the design values for these instruments and will not impact the ability of the in-core TCs to perform their design function. With respect to the in-core movable detectors, the full-power SPU flux levels will be higher. However, they are still within the design capability of the detectors. Therefore, the in-core TCs and movable detectors will continue to provide indication as designed.

2.4.1.2.4 Results

The changes to the instrumentation and controls for the SPU are the result of accident and transient analyses and system evaluations to verify the systems and controls will continue to provide the required indication, protection actions, and plant response as originally designed. The changes ensure the DNB values remain within acceptable limits and the RCS pressure boundary and the main steam pressure boundary are maintained within the design values. There are no new protection or control systems required to support the SPU. The identified instrumentation recalibration and instrument rescaling will ensure the instrumentation continues to allow monitoring plant process parameters during normal, transient, and accident conditions and provide protective functions as required.

2.4.1.3 Conclusions

Luminant Power has reviewed the evaluation related to the effects of the proposed SPU on the functional design of the reactor trip system, ESFAS, safe shutdown system, and control systems. Luminant Power concludes that the evaluation has adequately addressed the effects of the proposed SPU on these systems and that the changes that are necessary to achieve the proposed SPU are consistent with the plant's design basis. Luminant Power further concludes that the systems will continue to meet the requirements of 10 CFR 50.55a(a)(1), 10 CFR 50.55(a)(h), and GDCs 1, 4, 13, 19, 20, 21, 22, 23, and 24. Therefore, Luminant Power finds the proposed SPU acceptable with respect to instrumentation and controls.

Table 2.4.1.2-1 Steam Dump Setpoints – Loss of Load Controller					
Parameter	**Current Value**			**SPU**	
Loss of Load Controller	Unit 1 (Full Power T_{avg} > 583°F)	Unit 1 (Full Power T_{avg} < 583°F)	Unit 2	Units 1 and 2 (Full Power T_{avg} < 580°F)	Units 1 and 2 (Full Power T_{avg} ≥ 580°F)
Deadband (TC-500A)	5.0°F	3.0°F	5.0°F	2.0°F	5.0°F
Trip Open Setpoints					
Hi-1 ($T_{avg} - T_{ref}$) (TB-500B)	9.5°F	7°F	9.5°F	6.5°F	9.5°F
Hi-2 ($T_{avg} - T_{ref}$) (TB-500C)	14.0°F	11.0°F	14.0°F	11.0°F	14.0°F
Proportional Gain (TC-500A)	11.11%/°F	12.5%/°F	11.11%/°F	11.11%/°F	11.11%/°F

Table 2.4.1.2-2 Steam Dump Setpoints – Plant Trip Controller		
Parameter	**Current Value**	**SPU**
Plant Trip Controller	Units 1 and 2	Units 1 and 2
Deadband (TC-500D)	0.0°F	0.0°F
Trip Open Setpoints		
Hi-1 ($T_{avg} - T_{ref}$) (TB-500E)	15.6°F	20.0°F
Hi-2 ($T_{avg} - T_{ref}$) (TB-500F)	31.2°F	40.0°F
Proportional Gain (TC-500D)	3.2%/°F	2.5%/°F

2.4.2 Plant Operability/Component Sizing

2.4.2.1 Plant Operability

2.4.2.1.1 Regulatory Evaluation

Nuclear steam supply system (NSSS) instrumentation and control (I&C) systems are required to respond to the initiation of plant operational transients without initiating a reactor trip or engineered safety features (ESF) actuation signal. Luminant Power conducted an evaluation of the NSSS I&C systems response to design basis operational transients at stretch power uprate (SPU) conditions to ensure that the responses remain acceptable.

The acceptance criteria are based on:

- General Design Criterion (GDC) -13, insofar as it requires that I&C be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences (AOOs), and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the RCPB, the containment and its associated systems, and appropriate controls to maintain these variables and systems within prescribed operating ranges.

Current Licensing Basis

The Comanche Peak Nuclear Power Plant (CPNPP) Units 1 and 2 Final Safety Analysis Report (FSAR) Section 3.1 discusses the extent to which the design criteria for the plant structures, systems, and components (SSCs) important to safety comply with Title 10, Code of Federal Regulations, Part 50 (10 CFR Part 50), Appendix A, General Design Criteria (GDC) for Nuclear Power Plants. The adequacy of the CPNPP Units 1 and 2 designs relative to the GDC is discussed in FSAR Sections 3.1.1, 3.1.2, 3.1.3, 3.1.4, 3.1.5, and 3.1.6. Specifically, the adequacy of the CPNPP Units 1 and 2 I&C systems design was assessed by reviewing conformance to:

- GDC-13 is described in FSAR Section 3.1.2.4, General Design Criterion 13 - Instrumentation and Control.

  To ensure adequate safety, I&C systems are provided to monitor and control significant variables over their anticipated range for all conditions in the reactor core, reactor coolant system (RCS), steam and power conversion system, radioactive waste systems, ESF systems, and the Containment Building. Parameters that must be provided for operator use under normal operating and accident conditions are indicated in the Control Room in close proximity to the controls that maintain the indicated parameters in the proper range.

The quantity and types of process instrumentation provided ensures safe and orderly operation of all systems over the full design range of the plant. These systems are described in FSAR Chapters 6, 7, 8, 9, 10, 11, and 12.

FSAR Section 7.7 defines the general design objectives of the plant control systems as:

- Establishing and maintaining power equilibrium between primary and secondary systems during steady-state operation
- Constraining operational transients so as to preclude unit trip and to re-establish steady-state unit operation
- Providing the reactor operator with monitoring instrumentation that indicates all required input and output control parameters of the systems and provides the operator the capability of assuming manual system control

FSAR Section 7.7.2 discusses the plant operational transients that CPNPP Units 1 and 2 plant control systems must be able to sustain without initiating a reactor trip or an ESF actuation signal as follows:

- Step load increases or decreases of 10-percent load demand over the 15- to 100-percent power range without steam dump
- Ramp loading or unloading of 5-percent-per-minute load demand over the 15- to 100-percent power range without steam dump
- Step decrease of 50-percent loss of net load using steam dump
- Turbine trip with steam dump, when the plant is below the P-9 setpoint

2.4.2.1.2 Technical Evaluation

Analyses of the plant operational transients were performed using the proposed NSSS control system settings and setpoints to demonstrate adequate margin exists to relevant reactor trip and ESF actuation setpoints over the SPU full power T_{avg} normal operating range of 574.2° to 589.2°F. The SPU operating conditions are shown in Licensing Report (LR) Section 1.1, Nuclear Steam Supply System Parameters.

The following operational transients were addressed for the SPU:

- 5%/minute Unit Loading and Unloading (Condition I)
- 10% Step Load Increase (Condition I)
- 10% Step Load Decrease (Condition I)
- 50% Load Rejection (Condition I)
- Turbine Trip without a Reactor Trip from the P-9 Setpoint (Condition II)
- Normal Reactor Trip from Full Power (Condition II)

Input Parameters, Assumptions, and Acceptance Criteria

The key input parameters that are common to all the transients are summarized below. These inputs along with the assumptions noted below further ensure that there will be sufficient margin to the reactor trip and ESF actuation setpoints. The input parameters are:

- The plant operational transient analyses were performed at the high T_{avg} of 589.2°F and the Low T_{avg} of 574.2°F conditions. This analysis is then applicable to intermediate full-power T_{avg} values.
- In addition, a full-load T_{avg} of 580°F was analyzed for the turbine trip without a reactor trip transient and the 50-percent load rejection transients.
- The analyses are performed at 0- and 10-percent steam generator tube plugging (SGTP) conditions. Therefore, this analysis is applicable to intermediate SGTP levels.
- A 0.6-percent uncertainty allowance was applied to the initial power level operating conditions, for conservatism.
- Best-estimate reactor kinetics (for example moderator temperature coefficient (MTC), doppler power defect coefficients, control rod worth) at beginning of life (BOL) and hot full-power (HFP) conditions were modeled.
- The analysis used nominal steam dump valve capacities at the SPU conditions. The unit specific steam dump setpoints were used in the analyses.
- The Unit 1 analysis reflected the Δ76 steam generators and Unit 2 analysis reflected the D-5 steam generators.

The following assumptions were made for the plant operational transient analyses:

- NSSS control systems (rod control, steam dump control, feedwater/steam generator water level control, and pressurizer pressure control) are operating per design in the automatic mode of control. However, for the 10-percent step load decrease transient, the steam dump operation is not expected to be actuated for this transient. The feedwater/steam generator water level control system is approximated by equating feed flow to steam flow. LOFTRAN is used to predict the primary side transient responses for evaluating challenges to the primary side reactor trip setpoints, as well secondary side transient responses (that is, steam pressures, flow, and mass). The steam generator water level is indirectly modeled in LOFTRAN as the mass in the steam generator. LOFTRAN has been used in standard FSAR safety and operational transient analyses. Therefore, this code is acceptable to use for the purposes of the plant operability analyses. An evaluation was performed on the margin to the steam generator water level low-low reactor trip and high-high ESF actuation setpoints.

- The lead time constant on the low pressurizer pressure reactor trip signal is 2 second for CPNPP Unit 1.

- Plant parameters (that is, T_{avg}, pressurizer pressure, pressurizer level, and steam generator mass at the nominal water level) were assumed to be at the nominal full-power values with no uncertainty. These analyses are considered best estimate, which assumes that the plant is at the nominal operating conditions (that is, the plant parameters will be controlled to the target reference values). Without considering parameters uncertainties, the analysis results reflect what would be expected in the plant during and following a transient, which supports the objective of the analysis.

The following acceptance criteria are applicable for each of the transients analyzed:

- With the NSSS control systems assumed operational, there should be adequate operating margin to the relevant reactor trip and ESF actuation setpoints during and following the Condition I (normal operating) transients.

- Per NUREG-0737, Item II.K.3.10 (Reference 2), a turbine trip without a direct reactor trip below the P-9 setpoint should not challenge the pressurizer PORVs with the NSSS control systems operational.

- With the NSSS control systems assumed operational, a normal reactor trip transient from full power should not challenge the steam generator safety valves, low-low level heater cutoff setpoint, and the low pressurizer pressure ESF actuation setpoint.

- All control system responses should be smooth and stable without diverging oscillations.

Description of Analyses and Evaluations

The best-estimate analyses were performed using the LOFTRAN computer code. The analyses were performed using the multi-loop version of the Westinghouse LOFTRAN computer code. This computer model simulates the overall thermal-hydraulic and nuclear response of the NSSS as well as various control and protection systems. A LOFTRAN computer model was developed for CPNPP Units 1 and 2 at the SPU conditions. The LOFTRAN code has been reviewed and approved by the NRC (Reference 1). This code has been used to predict the plant responses for other SPU programs.

<u>10-Percent Step Load Decrease</u>

The 10-percent step load decrease transient results in an initial increase in the secondary side steam pressure and temperature as well as the primary side T_{avg} and pressure. The power mismatch between the turbine load and nuclear power and the resultant temperature error between T_{avg} and the reference temperature (T_{ref}) cause the rods to insert into the core, decreasing core power. The pressurizer pressure should not challenge the high pressurizer pressure reactor trip setpoint during and following a 10-percent step load decrease transient.

The 10-percent step load decrease transient was analyzed as a step decrease in turbine load from 100.6 percent (100 percent plus 0.6 percent) to 90-percent power.

10-Percent Step Load Increase

The 10-percent step load increase transient results in an initial decrease in the secondary side steam pressure and temperature as well as the primary side T_{avg} and pressure. The power mismatch between the turbine load and nuclear power and the resultant temperature error between T_{avg} and T_{ref} cause the rods to withdraw from the core, increasing core power. The 10-percent step load increase transient should not result in any automatic actuation of a reactor trip or ESF actuation signal. Therefore, the 10-percent step load increase transient verifies that there is adequate margin to the reactor trip and ESF actuation setpoints, primarily the high neutron flux reactor trip setpoint, low pressurizer pressure reactor trip, low steam line pressure ESF actuation setpoint, and the low-low level heater cutoff alarm setpoint.

The 10-percent step load increase transient was analyzed as a step increase in turbine load from 89.4 percent (90 percent minus 0.6 percent) to 100-percent power.

Turbine Trip Without Reactor Trip from P-9 Setpoint

After the Three Mile Island (TMI) incident, the NRC had expressed a concern on the implementation of any plant features that could increase the probability of a stuck-open pressurizer PORV. The NRC position is addressed in NUREG-0737, Item II.K.3.10 (Reference 2).

To satisfy the NRC position, a best-estimate analysis is performed to demonstrate that the pressurizer PORVs would not lift following a turbine trip without a reactor trip transient. Furthermore, it is desirable to prevent the steam generator steam pressure from challenging the first main steam generator safety valve.

A turbine trip without a reactor trip transient from the P-9 setpoint (50-percent power) was analyzed as a step decrease in turbine load from 50.6 percent (50 percent plus 0.6 percent) to 0-percent power.

50-Percent Load Rejection

The large load rejection transient results in an initial increase in the secondary side steam pressure and temperature as well as the primary side T_{avg} and pressure. The power mismatch between the turbine load and nuclear power and the resultant temperature error between T_{avg} and T_{ref} cause the rods to insert into the core, decreasing core power. The large load rejection transient (50-percent load rejection) with steam dump should not result in an automatic actuation of a reactor trip or ESF actuation signal. Therefore, the 50-percent load rejection transient verifies that there is adequate margin to the reactor trip or ESF actuation setpoints, primarily the overtemperature N16 (OTN16), overpower N16 (OPN16), and low pressurizer pressure reactor trip setpoints.

The 50-percent load rejection transient was analyzed as a step decrease in turbine load from 100.6 percent (100 percent plus 0.6 percent) to 50-percent power.

Normal Reactor Trip from Full Power

A normal reactor trip from full power transient was analyzed to verify that the steam generator safety valves are not challenged and that the pressurizer level does not shrink below the low-low level heater cutoff alarm setpoint to avoid uncovering the pressurizer heaters. Furthermore, the pressurizer pressure response should not challenge the low pressurizer pressure ESF actuation setpoint.

The normal reactor trip transient was analyzed from 100.6 percent (100 percent plus 0.6 percent) to 0-percent power.

Results

The results for the plant operability and margin to trip analysis for the CPNPP Units 1 and 2 SPU Program is provided below. The following summary and conclusions are valid for the full power T_{avg} window between 574.2°F and 589.2°F.

5-Percent-per-Minute Loading and Unloading

The 5-percent-per-minute loading and unloading transients are not limiting and are enveloped by the other transients analyzed. The 50-percent load rejection transient results in large pressure and temperature changes and is are more severe than the 5-percent-per-minute unloading transient. Similarly, the 10-percent step load increase is more severe than the 5-percent-per-minute loading transient. Thus, the 5-percent per minute unit loading and unloading transients are bounded by the 10-percent step load increase and 50-percent load rejection transient results, respectively, which were determined to be acceptable for the CPNPP Units 1 and 2 SPU conditions (see below).

10-Percent Step Load Decrease

The 10-percent step load decrease transient was analyzed, and the high pressurizer pressure reactor trip setpoint was not challenged. Therefore, the results show that the acceptance criteria are met at the CPNPP Units 1 and 2 SPU conditions.

10-Percent Step Load Increase

The 10-percent step load increase transient was analyzed, and the high neutron flux reactor trip, low pressurizer pressure reactor trip, low steam line pressure ESF actuation, and low-low level heater cutoff setpoints were not challenged. Therefore, the results show that the acceptance criteria are met at the CPNPP Units 1 and 2 SPU conditions.

Turbine Trip without Reactor Trip from P-9 Setpoint

Below the P-9 setpoint (50-percent reactor power), the automatic reactor trip on turbine trip is blocked. The turbine trip without a reactor trip transient from the P-9 setpoint satisfies the criteria of the NUREG-0737, Item II.K.3.10, that is, the pressurizer PORVs were not challenged for full-load T_{avg} values greater than or equal to 580°F with the current steam dump control system loss of load controller setpoints. For reduced full-load T_{avg} operating conditions less than 580°F, the revised loss of load controller setpoints need to be revised in order to satisfy the criteria of NUREG-0737, Item II.K.3.10 (see Table 2.4.1.2-1 of LR subsection 2.4.1). With the steam dump control system setpoints shown in Table 2.4.1.2-1 of LR subsection 2.4.1, the acceptance criterion is met at the CPNPP Units 1 and 2 SPU conditions.

50-Percent Load Rejection

The 50-percent load rejection transient was analyzed for full-load T_{avg} values greater than or equal to 580°F with the current steam dump control system loss of load controller setpoints. For reduced full-load T_{avg} operating conditions less than 580°F, the loss of load controller setpoints needed to be revised in order to satisfy the criteria of NUREG-0737, Item II.K.3.10 (see above). Therefore, the 50-percent load rejection transient was analyzed for reduced full-load T_{avg} operating conditions less than 580°F, with the revised loss of load controller setpoints (see Table 2.4.1.2-1 of LR subsection 2.4.1). The results showed that the OTN16, OPN16, and low pressurizer pressure reactor trip setpoints were not challenged.

The results show that the acceptance criteria are met at the CPNPP Units 1 and 2 SPU conditions.

Normal Reactor Trip Transient from Full Power

The normal reactor trip transient was analyzed with the current steam dump control system plant trip controller setpoints. The results showed that the low pressurizer pressure safety injection (SI) setpoint would be challenged for Units 1 and 2 at high T_{avg} conditions. Therefore, the plant trip controller setpoints were revised to ensure that SI would not actuate on low pressurizer pressure (see Table 2.4.1.2-2 of LR subsection 2.4.1). With these revised setpoints, the reactor trip transient showed that the low pressurizer pressure SI, steam generator safety valves, and the low-low level heater cutoff setpoints were not challenged.

Therefore, the results show that the acceptance criteria are met at the CPNPP Units 1 and 2 SPU conditions consistent with the plant trip controller setpoints as described in LR subsection 2.4.1.

Steam Generator Level Margin Evaluation

The steam generator level was not evaluated as part of the LOFTRAN analysis. Based on Westinghouse plant startup testing experience, the steam generator level shrink/swell will not challenge the low-low steam generator level reactor trip setpoint and the high-high steam

generator level ESF actuation setpoint provided that there is +/- 20-percent margin between the nominal control point and the low-low and high-high level setpoints.

The margin between the nominal control point (67-percent narrow range span (NRS) at Unit 1 and 64-percent NRS at Unit 2) and the low-low reactor trip setpoints (38-percent NRS at Unit 1 and 35.4-percent NRS at Unit 2) for CPNPP Units 1 and 2 is greater than 20 percent. Therefore, the steam generator level shrink on any normal operating transient should not challenge the reactor trip setpoint.

The margin between the nominal control point (67-percent NRS at Unit 1 and 64-percent NRS at Unit 2) and the high-high level ESF actuation setpoint (84-percent NRS at Unit 1 and 81.5-percent NRS at Unit 2) is 17 percent. This margin will be sufficient, since the plant settings are chosen to minimize the effects of steam generator level recovery.

2.4.2.1.3 Conclusions

Luminant Power has reviewed the effects of the proposed SPU on the plant capability of meeting its response to design basis operational transients. Luminant Power concludes that it has adequately addressed the effects of the proposed SPU on the plant operational capability and that the changes that are necessary to achieve satisfactory results at the SPU are consistent with the plant's design basis. Therefore, Luminant Power finds that, with appropriately revised control settings, the responses of the plant to operational transients at the proposed SPU are acceptable with respect to the plant capability of meeting its design basis operational transients and continuing to meet the current licensing basis with respect to the requirements as specified in GDC-13.

2.4.2.1.4 References

1. WCAP-7907, LOFTRAN Code Description, April 1984.

2. NUREG-0737, "Clarification of TMI Action Plan Requirements," Item II.K.3.10, Proposed Anticipatory Trip Modification, October 1980.

2.4.2.2 Pressurizer Pressure Control System Component Sizing

2.4.2.2.1 Regulatory Evaluation

The pressurizer pressure control system (consisting of the pressurizer heaters, spray valves, and power-operated relief valves (PORVs)) provides the means of controlling pressurizer pressure during steady-state operations and minimizes the pressurizer pressure variations during design basis operational transients. Nuclear Regulatory Commission (NRC) review standard RS-001, Revision 0 does not explicitly call out other guidance documentation for either current or post-uprate license basis reviews for pressurizer component sizing.

Luminant Power's review primarily focused on the effects of the proposed SPU on the Comanche Peak Nuclear Power Plant (CPNPP) Units 1 and 2 nuclear steam supply system (NSSS) pressurizer pressure control system components to ensure the system responses continue to meet their design basis operational functions for the operating conditions of the proposed SPU. Luminant Power's acceptance criteria for this review are based on 10 CFR Part 50.55a (a)(1); 10 CFR Part 50.55a(h); and:

- General Design Criterion (GDC) -1, insofar as it requires that structures, systems, and components (SSCs) important to safety be designed, fabricated, erected, constructed, and tested to quality standards commensurate with the importance of the safety functions to be performed.

- GDC-13, insofar as it requires that instrumentation be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to ensure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary (RCPB), and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

- GDC-19, insofar as it requires that a Control Room be provided from which actions can be taken to operate the nuclear unit safely under normal conditions, and maintain it in a safe condition under accident conditions, including LOCAs.

- GDC-24, insofar as it requires that the protection systems be separated from the control systems to the extent that a system satisfying all reliability, redundancy, and independence requirements of the protection systems is left intact in the event of a failure of any single control system component or channel, or failure or removal from service of any single control system component or channel that is common to the control and protection systems. Protection and control system interconnection will be limited to ensure that safety is not significantly impaired.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the adequacy of the CPNPP Units 1 and 2 pressurizer component sizing regarding conformance to:

- 10 CFR Part 50.55a is described in FSAR Section 5.2.1.1, "Compliance with 10 CFR Part 50.55a" and FSAR Table 5.2-1, "Applicable Code Addenda for Class 1 RCS Components."

- 10 CFR Part 50.55a(h) conformance is attained by meeting the plant's original license basis regarding compliance with Institute of Electrical and Electronics Engineers (IEEE) 279-1971.

- GDC-1, Quality Standards and Records, is described in FSAR Section 3.1.1.1.

  SSCs important to safety are designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Quality standards applicable to safety related SSCs are generally contained in codes such as the American Society of Mechanical Engineers Boiler and Pressure Vessel (ASME B&PV) Code. The applicability of these codes is specifically identified throughout the FSAR and is summarized in FSAR Section 3.2.5.

  FSAR Chapter 17 provides direct reference to the Quality Assurance Program established to provide assurance that safety related SSCs satisfactorily perform their intended safety functions. The procedures for generating and maintaining appropriate design, fabrication, erection, and testing records are contained within the referenced documents.

- GDC-13, Instrumentation and Control, is described in FSAR Section 3.1.2.13.

  Instrumentation and controls are provided to monitor and control neutron flux, control rod position, temperatures, pressures, flows, and levels as necessary to assure that adequate plant safety can be maintained. Instrumentation is provided in the RCS, steam and power conversion system, containment, engineered safety features systems, and other auxiliaries. Parameters that must be provided for operator use under normal operating and accident conditions are indicated in proximity with the controls for maintaining the indicated parameter in the proper range.

  The quantity and types of processing instrumentation provided ensures safe and orderly operation of all systems over the full design range of the plant. These systems are described in FSAR Chapters 6, 7, 8, 9, 11, and 12.

- GDC-19, Control Room, is described in FSAR Section 3.1.2.19.

  The Control Room provided is equipped to operate the unit safely under normal and accident conditions. Considering the detailed station design provisions to ensure continuous Control Room access, it is unlikely that the necessity could arise for evacuation of the Control Room. Nevertheless, provisions have been made to maintain the reactors in a safe hot standby condition if access to the Control Room is lost. Hot standby is maintained as described in FSAR Section 7.4.1.3.2. Furthermore, cold shutdown can be achieved from outside the Control Room through the use of suitable procedures as described in FSAR Section 7.4.1.3.3.

- GDC-24, Separation of Protection and Control Systems, is described in FSAR Section 3.1.2.24.

  The protection system is separate and distinct from the control systems. Control systems may be dependent on the protection system in that control signals are derived from protection system measurements, where applicable. These signals are transferred to the control system by isolation devices, which are classified as protection components. The adequacy of system isolation is verified by testing under conditions of postulated credible faults. The failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel that is common to the control and protection systems, leaves intact a system which satisfies the requirements of the protection system. Distinction between channel and train is made in this discussion. The removal of a train from service is allowed only during testing of the train. FSAR Chapter 7 gives further details.

Other FSAR sections that address the design features and functions of the pressurizer pressure control related systems or their control include Sections 5.2.2.11, 5.4.10, 5.4.13, and 7.7.1.5.

Licensing Report (LR) subsection 2.2.2.7 addresses the evaluation of the pressurizer and supports as pressure retaining components. LR subsection 2.5.2 addresses the pressurizer relief tank. LR subsection 2.8.4.2 evaluates overpressure protection at power. LR subsection 2.8.4.3 addresses overpressure protection during low-temperature operation.

2.4.2.2.2 Technical Evaluation

2.4.2.2.2.1 Input Parameters, Assumptions, and Acceptance Criteria

The NSSS pressurizer pressure control system components control the pressurizer pressure to the nominal value during steady-state operation and minimize the pressurizer pressure variations during design basis operability transients. A review of the installed capacities of the following NSSS pressurizer pressure control system components was performed for CPNPP Units 1 and 2 at the SPU conditions to ensure that these components continue to meet their design basis operational functions:

- Pressurizer PORVs

- Pressurizer spray valves
- Pressurizer heaters

The CPNPP Units 1 and 2 SPU conditions are shown in LR Section 1.1, Nuclear Steam Supply System Parameters. The analysis was performed to envelope the window of operating conditions, full-power normal operating T_{avg} of 574.2° to 589.2°F, at an uprated NSSS power of 3,628 MWt.

Input Parameters

The pressurizer pressure control system component sizing analysis was performed based on the following key input parameters:

- The plant is initially at nominal full-power T_{avg} plus a 6.0°F uncertainty. The analysis was performed at the lower normal operating limit for T_{avg} (574.2°F) and at the higher normal operating limit for T_{avg} (589.2°F).

- The initial pressurizer pressure is at nominal pressure of 2,235 psig (2,250 psia).

- The analyses were performed for 0-percent average steam generator tube plugging (SGTP) level because a minimum SGTP level corresponds to a higher steam pressure, which maximizes pressurizer insurge and pressure during heatup transients such as load rejections and load decreases. Therefore, the analysis at 0-percent SGTP bounds the 10-percent SGTP conditions.

- The initial pressurizer water level is at the nominal setpoint applicable to the full-power T_{avg} operating conditions. The pressurizer level program remains the same as the current program for full load T_{avg} values greater than 584.7°F. For the reduced T_{avg} range (below 584.7°F), the program was revised as part of the SPU (see LR Section 2.4.1 for more details).

- The capacity of the pressurizer PORVs is 210,000 lb/hour saturated steam per valve at 2,335 psig (2,350 psia). Units 1 and 2 each have two pressurizer PORVs. Therefore, each unit has a total PORV design capacity of 420,000 lb/hour.

- The capacity of the pressurizer spray valves is 450 gpm per valve. Units 1 and 2 each have two spray valves. Therefore, each unit has a total spray valve design capacity of 900 gpm.

- The rated capacities of the proportional and backup heaters are 400 and 1,400 kW, respectively, for Units 1 and 2. Therefore, each unit has a total heater design capacity of 1,800 kW.

- Best-estimate nuclear design parameters (moderator temperature coefficient, doppler-power defect, control rod worth, and boron concentration) at conservative beginning-of-life (BOL) conditions were used. BOL reactivity parameters have a lower differential rod worth and the least negative moderator temperature coefficient and using BOL parameters in the analysis yields more conservative results, which bound the full cycle of operation.

Assumptions

The pressurizer pressure control system component sizing analysis was performed based on the following key assumptions:

- The NSSS rod, pressurizer pressure, pressurizer level, feedwater/steam generator level, and steam dump control systems are assumed to be operational and functioning as designed in the automatic mode of control. Note that the pressurizer level and feedwater/steam generator level control systems were not explicitly modeled in this analysis. However, the pressurizer volume and steam generator mass were considered in the analysis.

- LOFTRAN is used to predict the primary side transient responses for evaluating challenges to the primary side reactor trip setpoints, as well secondary side transient responses (i.e., steam pressures, steam flow and mass); steam generator level is indirectly modeled in LOFTRAN as the mass in the generator. LOFTRAN has been used in standard FSAR safety and operational transient analyses; therefore, this code is acceptable to use for the purposes of the component sizing analysis.

Acceptance Criteria

The pressurizer pressure control system component sizing analysis was performed based on the following acceptance criteria:

- The installed capacity of the pressurizer PORVs should be adequate to prevent the pressurizer pressure from reaching the high pressurizer pressure reactor trip setpoint of 2,385 psig (2,400 psia) for the design basis large load rejection transient with steam dump actuation. Alternately, this sizing criterion for pressurizer PORVs is conservatively met if the total PORV capacity is greater than or equal to the peak pressurizer insurge flow rate during and following this transient.

- The installed capacity of the pressurizer spray valves should be adequate to prevent the pressurizer pressure from reaching the pressurizer PORV actuation setpoint of 2,335 psig (2,350 psia) for a 10-percent step-load decrease transient.

- The original sizing criterion for the pressurizer heaters was based on the physical dimensions of the pressurizer; that is, the design capacity of the pressurizer heaters should meet one kilowatt per one cubit foot (1 kW/ft^3) of the total pressurizer volume. The physical dimensions and the design capacity are not affected by the SPU.

Description of Analyses and Evaluations

The pressurizer PORVs and spray valves sizing analyses were performed using the LOFTRAN computer code. This computer code simulates the overall thermal-hydraulic and nuclear responses of the NSSS systems as well as the control and protection systems. A plant-specific LOFTRAN computer model was developed for CPNPP Units 1 and 2. The LOFTRAN code has been used to predict the plant responses for other SPU programs, and this methodology has been reviewed and approved by the NRC (Reference 1).

An evaluation was performed for the sizing of the pressurizer heaters (that is, no computer codes were used).

Pressurizer PORVs

The limiting transient that was analyzed to verify the installed capacity of the pressurizer PORVs is the design basis large-load rejection transient with steam dump. This transient provides the most severe RCS heatup. CPNPP Units 1 and 2 are designed to accept a large-load rejection of 50 percent from full power. Therefore, a 50-percent step-load decrease with steam dump transient was analyzed from 100 percent of full uprated NSSS power of 3,628 MWt to 50-percent power.

Pressurizer Spray Valves

The limiting transient that was analyzed to verify the installed capacity of the pressurizer spray valves is the design basis 10-percent step-load decrease transient. For a load change of 10 percent, the spray valves are the sole means of providing pressure control without actuating the pressurizer PORVs in the automatic mode of pressure control (that is, the steam dump control actuation is blocked). Therefore, a 10-percent step-load decrease transient was analyzed from 100 percent of full uprated NSSS power of 3,628 MWt to 90-percent power.

Pressurizer Heaters

The pressurizer heaters do not have a significant impact on the analyzed plant transients; the resulting pressurizer insurges/outsurges, and subsequent pressurizer pressure variations, are too rapid for the pressurizer heaters to influence. Therefore, the pressurizer heater capacity is considered acceptable for the SPU conditions.

Results

Pressurizer PORVs

For both CPNPP Units 1 and 2, the 50-percent load rejection transient results showed that the total installed capacity of the pressurizer PORVs (420,000 lb/hr) was sufficient to prevent the peak pressurizer pressure from challenging the high pressure reactor trip setpoint of 2,385 psig (2,400 psia) following a 50-percent load rejection transient at the SPU conditions. In addition, the peak pressurizer in-surge flow rate was verified to be less than the total installed capacity of

the pressurizer PORVs. Therefore, the installed capacity of the pressurizer PORVs is acceptable at the CPNPP Units 1 and 2 SPU conditions.

Pressurizer Spray Valves

For both CPNPP Units 1 and 2, the analysis results showed that the total installed capacity of the pressurizer spray valves (900 gpm) was sufficient to limit the peak pressurizer pressure to less than the pressurizer PORV actuation setpoint of 2,335 psig (2,350 psia) and avoid actuation of the PORVs during a 10-percent step-load decrease transient at the SPU conditions. Therefore, the installed capacity of the pressurizer spray valves is acceptable at the CPNPP Units 1 and 2 SPU conditions.

Pressurizer Heaters

The capacity of the pressurizer heaters is acceptable at the CPNPP Units 1 and 2 SPU conditions.

2.4.2.2.3 Conclusion

The effects of the SPU on the capacity of the NSSS pressurizer pressure control system components have been evaluated. The installed capacities of the pressurizer pressure control system components continue to meet the CPNPP Units 1 and 2 current licensing basis requirements with respect to 10 CFR 50.55a(a)(1) and 10 CFR 50.55(a)(h), GDCs-1, -13, -19, and -24. Therefore, Luminant Power finds the SPU acceptable with respect to the installed capacities of the pressurizer pressure control system components.

2.4.2.2.4 Reference

1. WCAP-7907, "LOFTRAN Code Description," April 1984.

2.5 PLANT SYSTEMS

2.5.1 Internal Hazards

2.5.1.1 Flooding

2.5.1.1.1 Flood Protection

2.5.1.1.1.1 Regulatory Evaluation

A review was conducted which addressed the areas of flood protection to ensure that safety-related structures, systems, and components (SSCs) are protected from flooding. This review covered flooding of safety-related SSCs from internal sources, such as those caused by failures of non-seismic tanks and vessels. The review focused on increases of fluid volumes in tanks and vessels assumed in the flooding analyses to assess the impact of any additional fluid on the flooding protection that is provided. Flood Protection under this section is related only to the following flood sources: (1) internal flooding to SSCs due to external events and (2) internal flooding due specifically to non-seismic tanks and vessels/equipment.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

The adequacy of the CPNPP flood protection system is assessed relative to conformance with the following:

- GDC-2, described in FSAR Section 3.1.1.2, Design Bases for Protection against Natural Phenomena.

- GDC-4, described in FSAR Section 3.1.1.4, Environmental and Dynamic Effects Design Bases.

Flood protection measures are discussed in FSAR Section 3.4, Water Level (Flood) Design.

Internal flooding from sources other than high energy line breaks and moderate energy line breaks, including internal flooding due to failure of non-seismic tanks and process equipment, is addressed in the following FSAR sections:

- FSAR 3.4.3, Flooding from Tank Rupture
- FSAR 3.11B, Environmental Design of Mechanical and Electrical Equipment

2.5.1.1.1.2 Technical Evaluation

The existing plant evaluations for flooding conditions inside containment, as well as in various other plant locations are reviewed to identify any changes related to the stretch power uprate (SPU). These current evaluations are summarized in FSAR Section 3.4.

As stated in FSAR Section 3.4.3 and Table 3.4-1, the effects of flooding in the Containment, Safeguards, Fuel and Auxiliary buildings from non-seismic Category 1 tanks on the shutdown capability of the plant were analyzed assuming that the release of tank contents is instantaneous and the sump pumps are not operational. In all cases, tank rupture results in local flooding only, with no effect on plant safety or the plant safe shutdown capability. As a result of SPU, non-seismic tank volumes will remain unchanged. Because there are no changes to the operating/design parameters which formed the basis for the current flooding evaluations, the performance of structures, systems, and components (SSCs) that are important to safety will remain acceptable and bounded for the SPU.

The existing flooding analysis for other plant locations is not affected by the SPU. The system parameters, which formed the basis of the existing flooding evaluations, do not change as a result of the SPU.

Additionally, internal flooding is addressed in the following Licensing Report (LR) sections:

- Internal flooding due to high energy line breaks and moderate energy line breaks in affected buildings, is addressed in LR subsection 2.5.1.3, Pipe Failures.
- Submergence inside containment is addressed in LR subsection 2.3.1, Environmental Qualification of Electrical Equipment.
- Protection of the Control building from flooding due to a break/leakage in the circulating water system, and protection from internal flooding in the turbine building and circulating water intake structure, is addressed in LR subsection 2.5.1.1.3, Circulating Water System.
- Internal flooding due to normal equipment leakage and floor drain system propagation is addressed in LR subsection 2.5.1.1.2, Equipment and Floor Drains.

2.5.1.1.1.3 Conclusion

Internal flooding from sources, other than high energy line breaks and moderate energy line breaks, from non-seismic tanks and vessels that could potentially affect safety-related components have not changed as a result of SPU. Therefore, it is concluded that safety-related SSCs will continue to be protected from flooding, and will continue to meet the CPNPP current licensing basis with respect to the requirements of GDCs -2 and -4 following implementation of the proposed SPU.

2.5.1.1.2 Equipment and Floor Drains

2.5.1.1.2.1 Regulatory Evaluation

The equipment and floor drainage system ensures that waste liquids, valve and pump leak offs, and tank drains are directed to the proper area for processing or disposal. The equipment and floor drainage system is designed to handle the volume of normal equipment leakage expected, prevent a backflow of water where credited that might result from maximum flood levels to areas of the plant containing safety-related equipment, and protect against the potential for inadvertent transfer of contaminated fluids to an uncontaminated drainage system. The review of the equipment and floor drainage system included the collection and disposal of liquid effluents outside containment. The review focused on any changes in fluid volumes or pump capacities that are necessary for the proposed SPU and are not consistent with previous assumptions with respect to floor drainage considerations. The Nuclear Regulatory Commission's (NRC's) acceptance criteria for the equipment and floor drainage system are based on General Design Criteria (GDC) -2 and -4 insofar as they require the equipment and floor drainage system to be designed to withstand the effects of earthquakes and to be compatible with the environmental conditions (flooding) associated with normal operation, maintenance, testing, and postulated accidents (pipe failures and tank ruptures).

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

The adequacy of the CPNPP flood protection system is assessed relative to conformance with the following:

- GDC-2, described in CPNPP FSAR Section 3.1.1.2, Design Bases for Protection against Natural Phenomena.

- GDC-4, described in CPNPP Peak FSAR Section 3.1.1.4, Environmental and Dynamic Effects Design Bases.

Functions and features of the equipment and floor drains systems regarding internal flooding are addressed in the following FSAR sections:

- FSAR 9.3.3, Equipment and Floor Drainage System
- FSAR 11.2, Liquid Waste Management System

2.5.1.1.2.2 Technical Evaluation

This evaluation addresses functions of the equipment and floor drains systems, including routing and control of leakage, and prevention of backflow of water/contaminated fluids to areas

of the plant containing safety-related equipment. Flooding caused by a high energy line break (feedwater line rupture), including discussion of impact on drainage for SPU conditions, and is addressed in LR subsection 2.5.1.3, Pipe Failures.

As addressed in FSAR Sections 9.3.3 and 11.2, the equipment and floor drain systems serve to route leakage from equipment and compartments in order to provide proper control of leakage, prevent uncontrolled communication between areas as necessary, and allow monitoring of leakage prior to disposition. The equipment and floor drains are included in the liquid waste disposal system.

As addressed in LR subsection 2.5.1.1.1.2, the SPU does not affect size or volume of fluid in the non-seismic tanks in plant areas where flooding from these tanks could affect safety-related components. Therefore, there is no additional leakage from these sources which could affect the equipment and floor drains systems.

The SPU does not affect the operating flow rates and pressures of the service water system, component cooling water system, fire water system, or residual heat removal system. Therefore, the SPU does not affect the capability of the floor drains systems to assist in prevention of flooding due to line breaks in these systems in applicable areas.

The function of the installed backflow prevention devices credited to prevent flooding of safety-related areas via backflow through floor drains is not affected by the SPU. No new areas requiring backflow prevention are required at SPU conditions.

As a result of the SPU, there will be no impact to the quantities of liquids that enter the equipment and floor drains systems from any of these sources. Flooding caused by a high energy line break (e.g., feedwater line rupture), including discussion of impact on drainage for SPU conditions, and is addressed in LR subsection 2.5.1.3, Pipe Failures.

2.5.1.1.2.3 Conclusion

The effects of the proposed SPU on the equipment and floor drainage system have been reviewed. There is no additional leakage from non-seismic tanks in areas containing safety-related equipment which would affect the equipment and floor drains systems. The SPU does not affect the capability of the floor drains systems to assist in prevention of flooding in applicable areas. The function of the installed backflow prevention devices credited to prevent flooding of safety-related areas via backflow through floor drains is not affected by the SPU, and no new areas requiring backflow prevention are required at SPU conditions.

Based on this, it is concluded that the equipment and floor drainage system will continue to meet the CPNPP current licensing basis with respect to the requirements of GDCs -2 and - 4 following implementation of the proposed SPU. Therefore, the proposed SPU is acceptable with respect to the equipment and floor drainage system.

2.5.1.1.3 Circulating Water System

2.5.1.1.3.1 Regulatory Evaluation

The circulating water system provides a continuous supply of cooling water to the main condenser to remove the heat rejected by the turbine cycle and auxiliary systems. This review of the circulating water system focused on changes in flooding analyses that are necessary due to increases in fluid volumes or installation of larger capacity pumps or piping needed to accommodate the proposed SPU. The acceptance criteria for the circulating water system is based on General Design Criterion (GDC) -4 for the effects of flooding of safety-related areas due to leakage from the circulating water system and the effects of malfunction or failure of a component or piping of the circulating water system on the functional performance capabilities of safety related structures, systems, and components (SSCs).

Current Licensing Basis

The adequacy of Comanche Peak Nuclear Power Plant (CPNPP) design relative to conformance to GDC-4 is addressed in Final Safety Analysis Report (FSAR) Section 3.1.1.4, Environmental and Dynamic Effects Design Bases.

Analyses/design features related to internal flooding due to leakage or a break in the circulating water system are addressed in the following FSAR sections:

- FSAR 3.4, Water Level (Flood) Design
- FSAR 9.2.1, Station Service Water System
- FSAR 10.4.5.3, Circulating Water System Safety Evaluation

2.5.1.1.3.2 Technical Evaluation

Evaluation of the impact of the SPU on analyses and design features related to internal flooding due to leakage or a break in the circulating water system is as follows: As discussed in Licensing Report (LR) subsection 2.5.8.1, the circulating water system flow rate and operating pressures do not change at SPU conditions. There are no modifications to the circulating water system resulting from the SPU.

Accordingly the analyses and design features related to internal flooding due to leakage or a break in the circulating water system for current plant conditions are unaffected by the SPU; protection of safety-related equipment continues to be provided.

2.5.1.1.3.3 Conclusion

Protection of safety related equipment from flooding due to a break or leakage in the circulating water system has been reviewed. Affected areas reviewed include the Turbine building, Control building, and the circulating water intake structure. It is concluded that, consistent with the CPNPP current licensing basis with respect to the requirements of GDC-4, since the circulating water system flow and operating pressures will remain unchanged for the SPU, and there are

no modifications to the Circulating Water System resulting from the SPU, the proposed SPU is acceptable with respect to the flooding from the Circulating Water System.

2.5.1.2 Missile Protection

2.5.1.2.1 Internally Generated Missiles

2.5.1.2.1.1 Regulatory Evaluation

The Luminant Power review concerned missiles that could result from in-plant component overspeed failures and high pressure system ruptures. The Luminant Power review of potential missile sources covered pressurized components and systems, and high-speed rotating machinery. The Luminant Power review was conducted to ensure that safety-related SSCs are adequately protected from internally generated missiles. In addition, for cases where safety-related SSCs are located in areas containing non-safety related SSCs, Luminant Power reviewed the non-safety related SSCs to ensure that their failure will not preclude the intended safety function of the safety related SSCs. The Luminant Power review focused on any increases in system pressures or component overspeed conditions that could result during plant operation, anticipated operational occurrences, or changes in existing system configurations such that missile barrier considerations could be affected.

The acceptance criteria for the protection of SSCs important to safety against the effects of internally generated missiles that may result from equipment failures are based on:

- GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with environmental conditions associated with normal operation, maintenance testing and postulated accidents.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

The adequacy of CPNPP design is assessed relative to conformance to:

- GDC-4, described in FSAR Section 3.1.1.4, Environmental and Dynamic Effects Design Bases (Criterion 4), which states that SSCs important to safety shall be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operating, maintenance, testing, and postulated accidents, including loss-of-coolant-accidents. These SSCs shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

SSCs important to safety are classified and designed in accordance with the codes and classifications indicated in FSAR Section 3.2.

Additional details that define the licensing basis for missile protection are described in FSAR Section 3.5, Missile Protection.

2.5.1.2.1.2 Technical Evaluation

2.5.1.2.1.2.1 Introduction

The basic approach to ensure missile protection of systems and components both inside and outside of containment involves:

- Examination of systems in order to identify and classify potential missiles
- Postulation of the generation of potential missiles and the provision of protection against them
- Verification of the design adequacy of equipment against the generation of missiles

Safety related SSCs at CPNPP are protected from internally generated missiles from sources inside and outside of containment. These missiles are generated by failures in high energy systems and the overspeeding of rotating components. FSAR Sections 3.5.1.1 and 3.5.1.2 discuss the selection of missiles, and missile protection for SSCs for internally generated missiles from sources inside and outside containment.

2.5.1.2.1.2.2 Description of Analyses and Evaluations

Missiles that are generated internally to the reactor facility (inside or outside containment) may cause damage to SSCs that are necessary for the safe shutdown of the reactor or for accident mitigation or may cause damage to the SSCs whose failure could result in a significant release of radioactivity. The potential sources of such missiles are valve bonnets and hardware retaining bolts, relief valve parts, instrument wells, pressure containing equipment (such as accumulators and high pressure bottles), high speed rotating machinery, and rotating components such as impellers and fan blades. FSAR Tables 3.5-1 and 3.5-6 identify internally generated missiles outside and inside containment respectively, and missile protection from these missiles.

The Luminant Power review focused on any increases in system pressures or component overspeed conditions due to the implementation of SPU that could result during plant operation, anticipated operational occurrences, or changes in existing system configurations such that missile barrier considerations could be affected.

The CPNPP SPU does not adversely impact the system pressures for the systems that could generate missiles. As such, the existing missile protection measures remain effective for SPU conditions. For plant areas containing safety-related SSCs, the SPU will not result in any

changes to existing missile sources or add any new components that could become a new potential missile source. The SPU will also not result in any system configuration changes that would impact any existing missile barrier considerations.

Refer to LR subsection 2.5.1.2.2, Turbine Generator, for evaluations of the impact of turbine missiles.

2.5.1.2.1.2.3 Results

The SPU does not adversely impact the pressures in the systems that could generate missiles. Hence, the existing missile protection measures remain effective for SPU conditions. For plant areas containing safety-related SSCs, the SPU will not result in any changes to existing missile sources or add any new components that could become a new potential missile source. The SPU will also not result in any system configuration changes that would impact any existing missile barrier considerations.

The results of the evaluations demonstrate that the SPU will not impact safety related SSCs with respect to internally generated missile concerns and will continue to meet the CPNPP Station current licensing basis with respect to the requirements of GDC-4.

2.5.1.2.1.3 Conclusion

The evaluation of the changes in system pressures and configurations that are required for the proposed SPU concludes that SSCs important to safety will continue to be protected from internally generated missiles following implementation of the proposed SPU and will continue to meet the current license basis with respect to GDC-4. Therefore, Luminant Power finds the proposed SPU acceptable with respect to internally generated missiles.

2.5.1.2.2 Turbine Generator

2.5.1.2.2.1 Regulatory Evaluation

The turbine control system, steam inlet stop and control valves, and low-pressure turbine steam inlet stop and control valves control the speed of the turbine under normal and abnormal conditions, and are thus related to the overall safe operation of the plants. The review of the turbine generator focused on the effects of the proposed SPU on the turbine overspeed protection features to ensure that a turbine overspeed condition above the design overspeed is not anticipated.

The acceptance criteria for the turbine generator is based on GDC-4, insofar as it relates to the protection of SSCs important to safety from the effects of turbine missiles by providing a turbine overspeed protection system (with suitable redundancy) to minimize the probability of generating turbine missiles.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

The adequacy of the CPNPP turbine generator design is assessed relative to conformance with the following:

- GDC 4, described in FSAR Section 3.1.1.4, Criterion 4 – Environmental and Dynamic Effects Design Bases, which states that SSCs important to safety are designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including a loss-of-coolant accident (LOCA). These SSCs are appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that can result from equipment failures and from events and conditions outside the nuclear power unit.

  Environmental conditions are described in FSAR Section 3.11, Environmental Design of Mechanical and Electrical Equipment.

2.5.1.2.2.2 Technical Evaluation

2.5.1.2.2.2.1 Introduction

The turbine generator and control system is described in FSAR Section 10.2.

The main turbine train is made up of one high-pressure turbine and two low-pressure turbines, all mounted on a common shaft. The steam flow path is first through the high-pressure turbine, through the moisture separators, then in a parallel path through the two low-pressure turbines. The main turbine operates at a design speed of 1,800 rpm. High-pressure steam is admitted to the high-pressure turbine through four main steam lines. Each main steam line has one stop valve and one control valve. These valves are controlled by the electro-hydraulic control (EHC) system. The stop and control valves are designed to shut off the steam flow to the turbine in the event the unit overspeeds beyond the setting of the overspeed trip. Following the steam flow through the moisture separator reheaters, steam enters the low-pressure turbines through four low-pressure stop valves and four low-pressure control valves (two steam lines per low-pressure turbine). These valves are also controlled by the EHC system. The purpose of the low-pressure stop and low-pressure control valves is to control the steam flow to the low-pressure turbines in the event of an overspeed condition. Installed in each extraction steam line from the high-pressure turbine are counter-weighted air-operated non-return check valves designed to prevent overspeeding of the turbine due to backflow of steam from the feedwater heaters following a turbine trip.

The overspeed protection system for the main turbine includes an electronic overspeed trip system. It is designed to trip the main turbine unit to ensure the turbine speed remains less than 120 percent of the design speed (2,160 rpm). There is also a turbine speed controller incorporated into the EHC system. This includes a load rejection circuit and redundant governor function. The load rejection circuit will rapidly close all control valves on a complete loss of load, and rapidly close the low pressure control valves on a partial loss of load.

2.5.1.2.2.2.2 Description of Analyses and Evaluations

In the event of a loss of electrical load on the turbine generator unit, the restraining torque on the turbine rotor unit is lost. The turbine control system is designed to close the control valves of both the high pressure turbine and the low pressure turbines. However, the steam energy entrapped in the turbine unit will cause the rotor to accelerate, potentially causing an overspeed condition.

The SPU increases the unit maximum power and the amount of entrapped energy. This results in an increase in expected peak overspeed.

The analysis was performed to demonstrate that the increase in power and entrapped steam energy at SPU conditions will not cause the turbine rotor to overspeed beyond the current design limit.

The following considerations were applied to this overspeed analysis:

- The normal operating turbine generator rotor "running" speed of 1,800 rpm will not change as the result of the SPU.
- The existing overspeed trip setpoint is 110 percent of design speed.
- The turbine design overspeed limit of 120 percent of design speed will not change as the result of the SPU.
- The electrical separation of the turbine generator from the grid was assumed to be at full-SPU load.
- The CPNPP Units 1 and 2 SPU involves the replacement of the high-pressure turbine rotors and inner casing steam path to accommodate the increase steam flows. The existing high-pressure turbine outer casing will remain unchanged.
- The existing CPNPP Units 1 and 2 SPU will not involve changes to the low-pressure turbine rotors, steam path, or casings.
- In addition to the steam volumes of the turbine block components (valves, cylinders, and cross-under/cross-over piping) the contribution of the steam and water volumes associated with the extraction lines and feedwater heaters were included in the overspeed analysis up to the non-return check valves (NRVs).

- There are no extraction steam or main steam piping changes planned as part of the SPU.
- The effect of the heaters in the condenser was not included in the stored volume as their contribution is considered insignificant to the overall results.

Based on previous overspeed analysis, the current overshoot is 8.4 percent above the 110-percent setpoint, resulting in an overspeed of approximately 118.4 percent. An evaluation was performed at the new SPU conditions that indicated that the overshoot will increase to 9.0 percent above the 110-percent setpoint, resulting in an overspeed of approximately 119 percent, which is less than the 120-percent design limit. Based on this value, there is no required change to the overspeed setpoint.

The operability and reliability of the turbine overspeed protection system is verified via the performance of routine turbine stop valve and control valve testing. The continued operability and reliability provided via valve performance testing at SPU conditions will be maintained.

The probability of turbine missile generation is analyzed via an evaluation of internal flaws (cracks) present in the low-pressure turbine discs and potential growth of the postulated flaws by a fatigue mechanism. The purpose of these analyses is to determine an acceptable low-pressure turbine external missile probability, thereby limiting the postulated flaw growth below the established failure threshold based upon a critical flaw size.

The CPNPP missile analysis is based on the following considerations:

- The analysis is limited to the evaluation of the low-pressure turbine discs. The most significant cause of a turbine missile is a burst-type failure of one or more bladed disks of a low-pressure turbine rotor. Failure of the high-pressure turbine rotor would be contained by relatively massive and strong turbine casings, therefore the high-pressure turbine rotor is bound by the low-pressure turbine missile analysis and is not considered in the disc inspection interval analysis. This is consistent with the original turbine disc integrity analysis performed. The basis for this position is that the high-pressure turbine is considered to be of an "integral" construction (single monoblock forging with axial or tangential entry blades). In addition, the material properties and the enhanced forging process associated with the replacement high-pressure rotor have provided an increase in the material toughness and therefore, a decrease in the likelihood of crack generation and growth.
- Favorable turbine-generator orientation with respect to the reactor containment buildings.
- The low-pressure turbine rotors, discs, and steam path are not being modified as part of the SPU implementation.
- The normal operating turbine generator rotor "running" speed of 1,800 rpm will not change as the result of SPU. The design overspeed of 120 percent will not change as the result of SPU.

- The methodology of the low-pressure disc integrity analysis is consistent with the current methodology, previously accepted by the Nuclear Regulatory Commission (NRC).

- The parameter that most directly impacts the inspection interval for the low-pressure turbine rotors is disc temperature. Slightly higher temperatures for some discs are expected under SPU conditions.

- SPU steam flow is bounded within the operating conditions applied.

The CPNPP missile analysis is based on the probabilistic method. The probability of an external missile (P1) is to be maintained below 1E-04. The value of P1 is composed of two terms:

- P_r = probability of an external missile for speeds up to and including 120-percent rated speed. This probability considers burst of the low-pressure rotor discs due to stress corrosion cracking.

- P_o = probability of an external missile for speeds greater than 120-percent rated speed. This probability, also referred to as the runaway overspeed probability, considers failure probability of the control and protection system.

One of the significant variables in the runaway overspeed probability determination is turbine valve test interval. Luminant Power plans to follow a semi-annual (26 weeks) turbine valve test interval. The turbine valve test analysis models the control and protection system used at CPNPP and determines the probability of a turbine-generator runaway overspeed (greater than 120-percent rated speed). This overspeed probability is used in the calculation of P1.

2.5.1.2.2.2.3 Results

Continued compliance with the turbine generator overspeed protection requirements was demonstrated at the SPU conditions with no plant changes. The SPU overspeed analyses results are as follows:

- The allowable overspeed trip setpoint will not need to be changed from its current set point of 110 percent. This will ensure that the turbine design overspeed limit of 120 percent will not be exceeded.

- For SPU conditions the probability of a missile remains less than 1E-04.

2.5.1.2.2.3 Conclusions

Luminant Power has reviewed the assessment of the effects of the proposed SPU on the turbine generator and concludes that the assessment has adequately accounted for the effects of changes in plant conditions on turbine overspeed. Luminant Power concludes that the turbine generator will continue to provide adequate turbine overspeed protection to minimize the probability of generating turbine missiles and will continue to meet the requirements of GDC-4

following implementation of the proposed SPU. Therefore, Luminant Power finds the proposed SPU acceptable with respect to the turbine generator.

2.5.1.3 Pipe Failures

2.5.1.3.1 Regulatory Evaluation

Luminant Power conducted a review of the plant design for protection from piping failures outside containment to ensure (1) such failures would not cause the loss of needed functions of safety-related systems and (2) the plant could be safely shut down in the event of such failures. Luminant Power review focused on the effects of pipe failures on plant environmental conditions, the Control Room habitability, and access to areas important to safe control of post accident operations where the consequences are not bounded by previous analyses. The acceptance criteria for pipe failures are based on General Design Criterion (GDC)-4, which requires, in part, that structures, systems, and components (SSCs) important to safety be designed to accommodate the dynamic effects of postulated pipe ruptures, including the effects of pipe whipping and discharging fluids.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the design criteria used during the licensing of Comanche Peak Nuclear Power Plant (CPNPP) are measured against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants.

The adequacy of the CPNPP pipe failure design is assessed relative to conformance with the following:

- GDC-4, described in FSAR Section 3.1.1.4, Criterion 4 – Environmental and Dynamic Effects Design Bases, which states that SSCs important to safety are designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents (LOCAs). Environmental conditions are described in FSAR Section 3.11, Environmental Design of Mechanical and Electrical Equipment.

  These SSCs are appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that can result from equipment failures and from events and conditions outside the nuclear power unit.

Other sections of the FSAR, which discuss pipe breaks outside containment, include Sections 3.6B.2 and 3.11N & B.

2.5.1.3.2 Technical Evaluation

2.5.1.3.2.1 Introduction

The pipe failure analysis identifies high and moderate energy piping system lines subject to failure. The analysis identifies the plant safety-related equipment potentially impacted by the postulated piping failures; determines the environmental effects resulting from the piping failures, and identifies the protection measures required to mitigate the effects of the piping failures. The environmental conditions resulting from this analysis are provided as input into the environmental qualification program. Refer to Licensing Report (LR) subsection 2.3.1, Equipment Qualification of Electrical Equipment. Refer also to LR subsection 2.2.1, Pipe Rupture Locations and Associated Dynamic Effects for the discussion of the impact of the SPU on pipe break locations.

The evaluation of pipe breaks outside containment considered the zones within the plant that contain systems required for safe shutdown and/or systems required to mitigate the effects of postulated pipe breaks.

Pipe failures inside containment are addressed in LR subsection 2.6.1 (Primary Containment Functional Design) and LR subsection 2.6.2 (Subcompartment Analysis)

2.5.1.3.2.2 Description of Analyses and Evaluations

The impact of the SPU on pipe failures was evaluated. The impact of the SPU on pipe whip, jet impingement, environmental conditions, and flooding was evaluated.

The evaluation of the impact of the SPU on the pipe failure analyses considers the events that are bounding with respect to the temperature and pressure conditions for each system containing essential equipment, that is, safety-related equipment required to operate for mitigation of the pipe failures.

2.5.1.3.2.2.1 High Energy/Moderate Energy Lines

The identification of the high energy and moderate energy lines does not change as a result of the SPU. The evaluations for the SPU conditions do not create any new or revised pipe break locations. Refer to LR subsection 2.2.1. No new equipment has been added that requires protection, and the existing high and moderate energy pipe break locations are not affected by implementation of the SPU.

2.5.1.3.2.2.2 Pipe Whip and Jet Impingement

The design of jet impingement shields and pipe rupture restraint protection features are based on the pipe break dynamic effects at design conditions. The evaluations performed did not identify any significant increases in operating conditions, except for the feedwater system, that would impact jet impingement and pipe whip analyses. Refer to LR Section 2.2.1.2.2 for

discussion of the impact of the increase in the feedwater operating conditions on jet impingement and pipe whip.

2.5.1.3.2.2.3 Evaluation of Piping Failures

Main Steam System

The main steam system operating conditions have changed for the SPU. The resultant mass and energy (M&E) releases for the limiting main steam line pipe failure have been revised. The revised mass and energy releases were evaluated for their impact on the area environmental conditions (compartment temperature and pressure).

Feedwater System

The feedwater system operating conditions have changed for the SPU. The current M&E releases utilized for Units 1 and 2 remain unchanged for the SPU since the feedwater operating conditions associated with the SPU are not significantly different from the current Units 1 and 2 conditions. The impact of flooding due to feedwater line failure was evaluated.

Auxiliary Feedwater System

There is no change to the design system pressure for uprate conditions (LR subsection 2.5.4.5). Therefore, there is no impact on the results of the pipe failure for the auxiliary feedwater system. The current M&E releases utilized for Units 1 and 2 remain unchanged for the SPU since the no load case is the limiting condition.

Auxiliary Steam System

There are no changes to the auxiliary steam system. Therefore there is no impact on the results of the pipe failure in this area.

Steam Generator Blowdown System

There are no changes to the design conditions of the steam generator blowdown system due to the SPU. Therefore, there is no impact on the results of the pipe failure in this system.

Chemical and Volume Control System

The SPU conditions do not change the design conditions of the chemical and volume control system. Therefore, there is no impact on the results of the pipe failure in this system.

Residual Heat Removal System

The RHR system is considered moderate energy outside containment. There are no changes to the design conditions of the residual heat removal system due to the SPU. Therefore, there is no impact on the results of the pipe failure in this system.

Safety Injection System

There are no changes to the design conditions of the safety injection system due to the SPU. Therefore, there is no impact on the results of the pipe failure in this system.

There are insignificant changes in the volumes of fluid or the M&E release for the above system break scenarios.

Moderate Energy Systems

There are no changes to the design conditions of moderate energy systems due to the SPU as described in FSAR Section 3.6B.2.5.3.

2.5.1.3.2.2.4 Summary of Impact on Building Environments

Main Steam System

The revised M&E release in the main steam penetration areas shows there is a small increase in temperature in the main steam and feedwater penetration areas outside of containment. The impact on the qualification of equipment as a result of the high energy line break in the main steam penetration areas is addressed in LR subsection 2.3.1.

Feedwater System

The flooding volume in the Safeguards Building is based on evaluating a non-mechanistic crack in the feedwater line equal in area to a one square-foot break. For Unit 1, the flooding volume was evaluated for the Model Δ76 steam generators and is still bounding for SPU. The flooding volume is still bounding for Unit 2 since the feedwater operating conditions associated with the SPU are not significantly different.

Flooding Outside Containment

The mass releases due to flooding outside containment are not impacted at the uprated conditions.

2.5.1.3.2.3 Results

The identification of the high energy and moderate energy lines does not change as a result of the SPU. The evaluations performed did not identify any significant increases in operating conditions that would impact jet impingement and pipe whip analyses. Therefore, the current design basis analyses and design adequacy for supports for pipe break, jet impingement, pipe whip, and effects due to moderate energy line break considerations remains valid for uprate conditions.

The impact of pipe failures for the main steam and feedwater lines was evaluated and the impact of the revised environmental conditions are addressed in LR subsection 2.3.1. There is no impact due to pipe failures of the remaining systems.

2.5.1.3.3 Conclusions

Luminant Power reviewed the changes that are necessary for the proposed SPU and the proposed operation of the plant, and concludes that SSCs important to safety will be protected from the dynamic effects of postulated piping failures in fluid systems and will meet the requirements of GDC-4 following implementation of the SPU. Therefore, the proposed SPU is acceptable with respect to protection against piping failures in fluid systems outside containment.

2.5.1.4 Fire Protection

2.5.1.4.1 Regulatory Evaluation

The purpose of the Fire Protection Program (FPP) is to provide assurance, through a defense-in-depth design, that a fire will not prevent the performance of necessary safe plant shutdown functions and will not significantly increase the risk of radioactive releases to the environment. The Luminant Power review focused on the effects of the increased decay heat on the plant's safe shutdown analysis to ensure that structures, systems, and components (SSCs) required for the safe shutdown of the plant are protected from the effects of the fire and will continue to be able to achieve and maintain safe shutdown following a fire. The acceptance criteria for the fire protection program are based on:

- 10 CFR 50.48 and associated Appendix R to 10 CFR 50, require the development of an FPP to ensure, among other things, the capability to safely shut down the plant.
- General Design Criterion (GDC) -3 requires that:
  - Safety-related SSCs be designed and located to minimize the probability and effect of fires.
  - Noncombustible and heat resistant materials are used.
  - Fire detection and fighting systems be provided and designed to minimize the adverse effects of fires on safety-related SSCs.
- GDC-5 requires that safety-related SSCs not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions.

Current Licensing Basis

FSAR Section 9.5.1.1 describes the Fire Protection Program for Units 1 and 2. The evaluation of fire hazards is included in the CPNPP Fire Protection Report (FPR) which follows the format of the U.S. Nuclear Regulatory Commission's (NRC's) "Supplementary Guidance on Information Needed for Fire Protection Program Evaluation" and the supplementary criteria in their September 30, 1976, letter.

The Comanche Peak Nuclear Power Plant (CPNPP) FPR Section I paragraph 2.5 "Summary" identifies that CPNPP is required to meet General Design Criteria 3 of 10 CFR 50 Appendix A and 10 CFR 50.48, sections (a) and (e). Appendix A to Branch Technical Position (BTP) APCSB 9.5-1 contains NRC staff guidance applicable to CPNPP.

The adequacy of the CPNPP design relative to the general design criteria is discussed in FSAR Sections 3.1.1.3 and 3.1.1.5. Specifically, the adequacy of CPNPP design relative to:

- GDC-3 is described in FSAR Section 3.1.1.3, Criterion 3 – Fire Protection. As described in this FSAR Section, fire detection and fighting systems of appropriate capacity and capability are provided to minimize the adverse effects of fire on SSCs important to safety.

- GDC-5 is described in FSAR Section 3.1.1.5, Criterion 5 – Sharing of Structures, Systems, and Components. As stated in this section, sharing of systems and components does not impair system capability to perform safety functions, e.g., the orderly shutdown and cooldown of one unit in the even of an accident in the other unit.

2.5.1.4.2 Technical Evaluation

2.5.1.4.2.1 Introduction

Fire Protection Program

The CPNPP fire protection program is described in both FSAR 9.5.1 and the Fire Protection Report.

The overall Fire Protection Program was developed utilizing the defense in depth concept. This concept is a combination of:

1. Preventing fires from starting

2. Quickly detecting and suppressing fires that do occur to limit the extent of damage

3. Designing plant safety systems so that a fire that becomes fully established and burns for a considerable time, in spite of the fire protection systems provided, will not prevent essential plant safety functions from being performed.

The FPR consolidates a detailed summary of the CPNPP regulatory-required Fire Protection Program into a single document, and, as such, embodies the fire protection program. The report documents CPNPP's Fire Protection Plan, Fire Hazards Analysis, and Fire Safe Shutdown Analysis. The FPR also includes requirements for fire protection equipment/systems to assure they are operable and properly maintained.

The program description/evaluation in this section addresses:

- Fire Hazards Analysis Report (FHAR)
- Fire Safe Shutdown Analysis Report (FSSAR)
- Administrative requirements for fire protection system and equipment at CPNPP.

Fire Protection

Administrative controls, personnel requirements for fire prevention and manual fire suppression activities, and fire protection systems and features, including fire detection and automatic and manually operated suppression systems are discussed in the FPR and FSAR 9.5.1. The administrative controls include controls to minimize the amounts of combustibles to which a safety-related/safe shutdown area may be exposed, control of hot work, impairment monitoring, etc.

Fire Hazards Analysis Report

The FHAR as included in the FPR includes the fire hazards analysis for each fire area within Unit 1, Unit 2, and the common buildings that contain equipment and components necessary to fulfill fire safe shutdown functions of the Unit 1 and/or Unit 2 reactor.

Fire protection features are also described in the FHAR including passive fire protection features, active fire protection features, manual FPR, reactor coolant pump lube oil collection system, emergency lighting, communications, and National Fire Protection Association (NFPA) Code compliance deviations.

Fire Safe Shutdown Analysis

The FSSAR as included in the FPR presents the analysis of the fire safe shutdown capability for CPNPP in accordance with the guidelines outlined in Appendix A to BTP APCSB 9.5.1, "Guidelines for Fire Protection for Nuclear Power Plants" docketed prior to July 1, 1976.

This report includes the regulatory criteria and operating license commitments pertaining to the fire safe shutdown analysis (FSSA), the analysis bases and supporting information, the CPNPP plant configuration for fire safe shutdown, the analysis methodology, and a summary of the important analysis results that established the design for CPNPP Fire Safe Shutdown.

The FSSA considers the effects of fire on plant equipment and identifies methods for achieving safe shutdown. The fundamental basis for this analysis is that a single fire occurs in any plant area coincident with a loss-of off-site power. Equipment normally present in the plant is

assumed to be functional at design capability and available for use in fire safe shutdown and may be lost only as a result of fire damage or the loss-of off-site power. Other external events, accidents, or failures are not assumed to occur currently with the postulated fire or any subsequent activities to achieve cold shutdown conditions within 72 hours.

2.5.1.4.2.2 Description of Analyses and Evaluations

Fire Protection

Apart from plant modifications, the SPU does not affect the following elements of the fire protection program:

- Addition of new combustible material
- Fire barriers, penetrations, doors, or the plant communication system
- Ventilation air flow patterns
- Plant fire programs or the Fire Protection Program Report
- Fire wraps and fire coatings on structural steel
- Fire protection suppression or fire detection system components
- Safety-related components within an area protected by the fire suppression system

Plant modifications required in support of the SPU will be reviewed to ensure any design changes do not adversely impact existing Fire Protection Program requirements.

Safe Shutdown Analysis

Safe Shutdown Systems/Components

Plant modifications required in support of the SPU will be reviewed to ensure any design changes do not adversely impact BTP APCSB 9.5-1 Appendix A requirements.

Alternative Shutdown Capability

Analyses were performed to demonstrate that the plant can be cooled down from normal operating temperature to cold shutdown at SPU conditions.

- An analysis shows that the plant can be cooled from normal operating temperature to residual heat removal system initiation conditions with two available steam generators within 13.5 hours after reactor shutdown.

- An analysis shows that the plant can be cooled from residual heat removal system initiation conditions to cold shutdown within 72 hours after reactor shutdown, assuming single train cooldown and residual heat removal cooldown start time of 13.5 hours after reactor shutdown.

Based on the above analyses, the requirement to achieve cold shutdown conditions within 72 hours after reactor shutdown continues to be met for SPU conditions.

Time-Critical Tasks

Time critical tasks are identified in the thermal/hydraulic analysis of the fire safe shutdown scenario. Operations procedures implement the time critical tasks to:

- Transfer PORV control to hot shutdown panel within five minutes
- Establish seal return flow within 30 minutes
- Start plant cooldown prior to two hours or pressurizer level exceeding 92 percent

From a review of the thermal/hydraulic analysis calculation, SPU does not impact these time critical operator actions. The corresponding plant procedures show that CPNPP can be shut down and cooled down to cold shutdown conditions within 72 hours under SPU conditions.

Safe Shutdown Procedures

Operations abnormal procedures have been written to respond to a fire in the Control Room or Cable Spreading Room. These operations procedures provide the necessary instructions to shut down and cool down the plant in response to a fire safe shutdown scenario. These procedures demonstrate that the plant can be shutdown and cooled down to cold shutdown conditions within 72 hours under SPU conditions.

The increase in decay heat due to the SPU does not result in an increase in the probability of a radiological release resulting from a fire based on adequate alternative shutdown capability, timely implementation of critical tasks, and adequate safe shutdown procedures as identified above.

2.5.1.4.3 Conclusion

The evaluation of the fire-related safe shutdown assessment concludes that the effects of the increased decay heat on the ability of the required systems to achieve and maintain safe shutdown conditions have been adequately addressed. It is further concluded that the CPNPP Station Fire Protection Program will continue to meet the requirements of 10 CFR 50.48, Appendix A to BTP APCSB 9.5-1, and will continue to meet the CPNPP current licensing basis with respect to the requirements of GDC-3 following implementation of the proposed SPU. Therefore, the proposed SPU is acceptable with respect to fire protection.

2.5.2 Pressurizer Relief Tank

2.5.2.1 Regulatory Evaluation

The pressurizer relief tank (PRT) is a pressure vessel provided to condense and cool the discharge from the pressurizer safety and relief valves. The tank is designed with a capacity to absorb discharge fluid from the pressurizer relief valve during a specified step-load decrease. The PRT system is not safety related and is not designed to accept a continuous discharge from the pressurizer. The PRT was reviewed to ensure that operation of the tank is consistent with transient analyses of related systems at the stretch power uprate (SPU) power level and that failure or malfunction of the PRT system will not adversely impact safety related structures, systems, and components (SSCs).

The review focused on any design changes related to the PRT and connected piping, and changes related to operational assumptions that are necessary in support of the proposed SPU that are not bounded by previous analyses.

The review showed that:

- The steam condensing capacity of the tank and the tank rupture disc relief capacity are adequate, taking into consideration the capacity of the pressurizer power-operated relief valves (PORVs) and safety valves.
- The piping to the tank is adequately sized.
- Systems inside containment are adequately protected from the effects of high-energy line breaks and moderate-energy line cracks in the pressurizer relief system.

The acceptance criteria for this review are:

- General Design Criterion (GDC)-2, insofar as it requires that SSCs important to safety be designed to withstand the effects of earthquakes.
- GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate and be compatible with specified environmental conditions, and be appropriately protected against dynamic effects, including the effects of missiles.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the adequacy of the CPNPP design relative to:

- GDC-2, Design Bases for Protection Against Natural Phenomena, is described in FSAR Section 3.1.1.2.

  Those features of plant facilities that are essential to the prevention of accidents that could affect the public health and safety or to the mitigation of accident consequences are designed to:

  1. Quality standards that reflect the importance of the function to be performed. Approved design codes are used when appropriate to the nuclear application.

  2. Performance standards that enable the facility to withstand, without loss of the capability to protect the public, the appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, and appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena.

- GDC-4, Environmental and Dynamic Effects Design Bases, is described in FSAR Section 3.1.1.4.

  SSCs important to safety are designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operating, maintenance, testing, and postulated accidents including LOCAs. These items are appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids that can result from equipment failures and from events and conditions outside the nuclear power unit.

2.5.2.2 Technical Evaluation

2.5.2.2.1 Introduction

The PRT system is described in FSAR Section 5.4.11. The steam and water discharged from the various safety and relief valves inside the containment is routed to the PRT if the discharged fluid is of reactor grade quality.

The tank normally contains water and a predominantly nitrogen atmosphere. In order to obtain effective condensing and cooling of the discharged steam, the tank is installed horizontally and the steam is discharged through a sparger pipe located near the bottom, under the water level. The sparger holes are designed to ensure a resultant steam velocity close to sonic.

The tank is also equipped with an internal spray and a drain that are used to cool the water following a discharge. Cold water is drawn from the reactor makeup water system, or the

content of the tank is circulated through the reactor coolant drain tank heat exchanger of the Waste Processing System and back into the spray header.

The nitrogen gas blanket is used to control the atmosphere in the tank and to allow room for the expansion of the original water plus the condensed steam discharge. The tank gas volume is calculated using a final pressure based on a design pressure of 100 psig. The design discharges raises the worst case initial conditions to 50 psig, a pressure low enough to prevent fatigue of the rupture discs. Provisions are made to permit the gas in the tank to be periodically analyzed to monitor the concentration of hydrogen and/or oxygen.

The contents of the vessel can be drained to the waste holdup tank in the Waste Processing System or the recycle holdup tank in the boron recycle system via the reactor coolant drain tank pumps in the Waste Processing System.

2.5.2.2.2 Description of Analyses and Evaluations

An assessment of the PRT was performed to ensure that the tank is capable of performing its intended function for the range of NSSS design parameters (Licensing Report (LR) Section 1.1).

The pressurizer safety valves require an adequate capacity to ensure that the RCS pressure does not exceed 110 percent of system design pressure. This is the maximum pressure allowed by the ASME Code, Section III, NB-7300 and NC-7300. RCS design pressure has not changed for the SPU. Based on the range of NSSS design parameters for the SPU, and analysis of the loss of external electrical load transient was performed. The analysis results confirmed that the installed pressurizer safety valves capacity is adequate to preclude RCS overpressurization. Based on the analysis results, the pressurizer surge line, safety valve inlet piping, and safety valve discharge piping (including the PRT sparger pipe) designs are also adequate, since they are based on safety valve design capacity.

The PRT design is based on the total safety valve capacity are conservatively sized to condense and cool a steam discharge equal to 105 percent of the full-power pressurizer steam volume. The amount of energy absorbed by the PRT is related to the volume and pressure of the discharge steam. The loss of external electrical load transient analysis determined that the pressurizer steam mass and energy discharged into the PRT is less than the design bases discharge – the PRT design remains conservative. The mass of coolant maintained in the PRT exceeds the mass of coolant required to condense the worst-case steam discharge associated with a postulated loss of external load transient at SPU conditions.

The PORVs are required to have adequate capacity to prevent a pressurizer high-pressure reactor trip for an external load reduction of up to 50 percent of rated electrical load. Based on the range of NSSS design parameters for the SPU, an analysis was performed (LR subsection 2.4.2.2) which confirmed that the installed PORVs capacity is adequate to preclude a pressurizer high pressure reactor trip. Based on these results, the PORVs inlet and discharge piping design is adequate, since the piping design is based on the PORVs design capacity. The mass and energy addition to the PRT during load rejection is not limiting with respect to the

PRT design, since this transient discharge is less severe than the loss of external electrical load transient discharge.

The PRT high and low level alarm setpoints ensure adequate coolant is maintained in the tank to condense and cool the design bases discharge, and to prevent the PRT temperature and pressure from exceeding the criteria of 200°F and 50 psig, respectively.

2.5.2.2.3 Results

The current PRT design bases bound the SPU loss of external load analysis mass and energy addition, such that the PRT continues to meet its design basis mass and energy addition.

The mass of coolant maintained in the PRT exceeds the mass of coolant required to condense the worst-case steam discharge associated with a postulated loss of external load transient at SPU conditions.

The current design basis for PRT interface support function is also not impacted. These support functions include primary grade water makeup for cooling, nitrogen for pressure control, gas analyzer connection for periodic sampling, and PRT vent and drain.

The SPU does not have any adverse effects on the PRT. The pressurizer safety valves are adequate for the SPU conditions since the RCS design pressure does not change as a result of the SPU. Also, since the design of the PRT is conservative, the SPU does not impact the design basis discharge.

2.5.2.3 Conclusions

As a result of the SPU, the PRT will operate in a manner consistent with the design requirements and will be protected in accordance with GDC-2 and -4 with respect to the current CPNPP licensing basis.

2.5.3 Fission Product Control

2.5.3.1 Fission Product Control System and Structures

2.5.3.1.1 Regulatory Evaluation

The Luminant Power review of fission product control systems and structures covered the basis for developing the mathematical model for design basis loss-of-coolant-accident (LOCA) (DBLOCA) dose computations, the values of key parameters, the applicability of important modeling assumptions, and the functional capability of ventilation systems used to mitigate the consequences of releases.

The Luminant Power review primarily focused on any adverse effects that the proposed stretch power uprate (SPU) may have on the assumptions used in the analyses for control of fission products. The Nuclear Regulatory Commission's (NRC's) acceptance criteria are based on:

- General Design Criterion (GDC) -41, insofar as it requires that the containment atmosphere cleanup system be provided to reduce the concentration of fission products released to the environment following postulated accidents.

- In addition, GDC-42 and -43 are also assessed.

Following the postulated design basis accident (DBA), engineered safety feature (ESF) fission-product removal and control systems are required to perform a safety-related function as described in this section.

GDC-19 of Appendix A to 10 CFR Part 50 and 10 CFR 100 require adequate radiation protection during access to and occupancy of the Control Room and control of radioactive releases to the environment. The secondary atmosphere cleanup systems satisfying GDC-19 and 10 CFR 100 are described in Final Safety Analysis Report (FSAR) Section 6.5.1.

GDC-41, -42, and -43 of Appendix A to 10 CFR Part 50 require that containment atmosphere cleanup systems be provided as necessary to reduce the amount of radioactive material released to the environment following a postulated DBA and that these systems be designed to permit appropriate periodic inspection and testing to ensure their integrity, capability, and operability. For this purpose, the containment spray system (CT) and the containment spray chemical additive subsystem are included as an ESF in the design of CPNPP Units 1 and 2 and are described in FSAR Section 6.5.2.

As required in GDC-19 of Appendix A to 10 CFR Part 50, the Control Room filtration units are provided to ensure a safe environment and to permit access and occupancy of the Control Room during and after a DBA. FSAR Section 9.4.5 provides the system mode of operation of the ESF ventilation system. These systems support the current dose consequences provided in FSAR Sections 15.1.5 (MSLB), 15.3.3 (Locked Rotor), 15.4.8 (Rod Ejection), 15.6.2 (Break in an Instrument Line), 15.6.3 (SGTR), 15.6.5 (LOCA), and 15.7 (Radiological Releases from a Subsystem or Component).

2.5.3.1.2 Technical Evaluation

Each emergency filtration unit is designed to handle approximately 16 percent of the required airflow supplied within the Control Room envelope. This percentage is based on an air change rate. The emergency recirculation airflow rate allows that Control Room volume to be filtered approximately once every hour. Control room area ventilation system operation is explained in FSAR Section 9.4.1.

High efficiency particulate absorption (HEPA) filters and iodine adsorbers used in these filter trains comply with the construction and efficiency requirements of NRC Regulatory Guide 1.52 as described in the FSAR.

The Control Room atmosphere cleanup systems conform to the criteria established in the regulatory positions of NRC Regulatory Guide 1.52 as described in the FSAR. Filters are designed to seismic Category I requirements. Redundancy of equipment and power supplies enables the systems to sustain a single, active failure without loss of function during a LOCA, loss of offsite power, and normal plant operation.

The CT has the dual function of removing heat as well as fission-product iodine from the containment atmosphere. Equipment descriptions and principal design parameters for those system components required for the heat removal function as delineated in FSAR Section 6.2.2.

Containment atmosphere purification and cleanup are accomplished by the CT. A chemical additive subsystem is provided to inject sodium hydroxide into the spray water. The sodium hydroxide maintains the pH of the spray water at an acceptable value for fission product removal from the post-LOCA containment atmosphere.

In addition, Control Room is pressurized with respect to the environment to prevent infiltration of unfiltered and unmonitored air and to account for leakages, as specified in LR Section 2.6.4.

Results/Acceptance Criteria

The acceptance criteria for these systems are that the offsite and Control Room doses meet regulatory requirements.

The effect of the SPU is an increase in source term, which is considered in the new LOCA dose analysis discussed in LR Section 2.9. A review of this section indicates that the ventilation and filtration systems and CT, in conjunction with other SSCs, are effective in limiting both Control Room and offsite doses to within regulatory limits.

2.5.3.1.3 Conclusions

Luminant Power has performed an assessment of the effects of the proposed SPU on fission product control systems and structures. The increase in fission products and changes in expected environmental conditions that would result from the proposed SPU have been accounted for. Luminant Power further concludes that the fission product control systems and structures will continue to provide adequate fission product removal in post-accident

environments following implementation of the proposed SPU. Based on this, Luminant Power concludes that the fission product control systems and structures will continue to meet the current licensing basis with respect to the requirements of GDC-41. Therefore, Luminant Power finds the proposed SPU acceptable with respect to the fission product control systems and structures.

2.5.3.2 Main Condenser Evacuation System

2.5.3.2.1 Regulatory Evaluation

The main condenser evacuation (CV) system consists of three condenser exhausting vacuum pumps that perform the following:

- Initially establish main condenser vacuum (hogging)
- Maintain condenser vacuum once it has been established (holding)

The review focused on the effects of the proposed SPU on the system's capability to maintain condenser vacuum, modifications to the system that may affect gaseous radioactive material handling and release assumptions, and design features to preclude the possibility of an explosion (if the potential for explosive mixtures exists). The Nuclear Regulatory Commission's (NRC's) acceptance criteria for the main condenser evacuation system are based on:

- General Design Criterion (GDC) -60, insofar as it requires that the plant design include means to control the release of radioactive effluents.

- GDC-64, insofar as it requires that means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences and postulated accidents.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

The adequacy of the CPNPP design relative to conformance to:

- GDC-60 is described in FSAR Section 3.1.6.1, Criterion 60 – Control of Releases of Radioactive Materials to the Environment. As described in this FSAR section, the handling, control, and release of radioactive materials are in compliance with 10 CFR 50, Appendix I.

  Radioactive gaseous waste effluent activity levels are monitored prior to release through the plant vent. Under conditions of coincident fuel failure and steam generator tube leakage, the condenser exhaust vacuum pump discharge gases are cascaded into a

common discharge line which is directed to the Primary Plant Ventilation System for controlled release to the atmosphere through an exhaust unit and the plant vent. FSAR Section 9.4 describes the Primary Plant Ventilation System and non-ESF exhaust units which satisfy GDC-60. Chapter 11 describes the radioactive waste processing systems' design criteria, holdup capacities, and estimated releases of radioactive effluents to the environment.

- GDC-64 is described in the FSAR Section 3.1.6.5, Criterion 64 – Monitoring Radioactivity Releases.

  Radioactivity levels contained in the facility effluent discharge paths and in the plant environs are continuously monitored during normal and accident conditions by the Process Radiation Monitoring System described in FSAR Section 11.5. A radiation monitor is provided on the condenser exhaust vacuum pumps discharge line to continuously measure the radiation level of the condenser off-gases discharged to the atmosphere with indication given in the Control Room. Provisions for monitoring the plant areas for radioactivity are included in the Area Radiation Monitoring System described in FSAR Section 12.3.4. In addition to the installed detectors, periodic surveys are conducted using portable equipment as discussed in FSAR Section 12.5.

2.5.3.2.2 Technical Evaluation

2.5.3.2.2.1 Introduction

The condenser evacuation system is discussed in the FSAR Section 10.4.2. The condenser evacuation system removes non-condensible gases from the condenser to draw a vacuum for startup and then to help maintain condenser vacuum during normal operation. The condenser evacuation system contains three 100-percent capacity, motor-driven, two-stage, rotary-type condenser exhausting vacuum pumps. The two main and two auxiliary condenser shells share a common header connection to the pumps.

2.5.3.2.2.2 Description of Analyses and Evaluations

The condenser air removal system must be capable of removing non-condensible gases and air in-leakage from the condenser shells (steam space) to maintain vacuum. Air in-leakage will not be affected by the SPU since it is entirely due to the physical design of the condenser and its state of degradation. In addition, any existing air in-leakage may be slightly reduced due to the higher condenser backpressure at SPU. Therefore, the condenser evacuation system is evaluated by comparing its pre-SPU removal capability with its removal capability based on the expected increase in non-condensible gases resulting from the increased low pressure turbine exhaust flow rate at SPU conditions. Refer to Licensing Report (LR) subsection 2.5.5.2, Main Condenser, for additional discussion related to the condenser.

Refer to LR subsection 2.10.1, Occupational and Public Radiation Doses for the evaluation of plant radioactive monitoring and control of releases of radioactive materials to the environment

in compliance of GDC-60 and -64. For gaseous radioactive material handling, refer to LR subsection 2.5.6.1, Gaseous Waste Management System.

Since there is no potential for explosive mixtures in the condenser, it was not included in the evaluation.

2.5.3.2.2.3 Results

The condenser evacuation system has been evaluated by comparing its pre-SPU removal capability with its removal capability based on the expected increase in non-condensible gases resulting from the increased low pressure turbine exhaust flow rate at SPU conditions. At a condenser pressure of approximately 10 in. Hg abs., the condenser exhaust vacuum pumps automatically switch from the hogging to the holding mode of operation. During the holding sequence, the number of condenser exhaust vacuum pumps in operation can be reduced to less than three depending on the load and the circulating water temperature.

The pre-SPU capacity of the condenser exhausting vacuum pumps was compared to the Heat Exchange Institute (HEI) standards venting equipment capacities required for the SPU steam flow rate. The requirements for the condenser evacuation system at SPU conditions are bounded by the current design.

The condenser exhausting vacuum pumps also evacuate non-condensible gases from the condenser during startup. Since startup conditions do not change due to SPU operation, the condenser exhausting vacuum pumps are adequate at SPU conditions.

The design of the main condenser evacuation system does not change following the implementation of the SPU. Therefore, the SPU does not impact the ability of Comanche Peak regarding the control of radioactive material or the monitoring of releases in accordance with GDC-60 and -64, respectively. Related to the impact of SPU on radiological effluent releases from Comanche Peak and compliance with 10 CFR 50, Appendix I, refer to Section 2.10.1, Occupational and Public Radiation Doses.

2.5.3.2.3 Conclusion

The evaluation concluded that the main condenser evacuation system will maintain its ability to remove non-condensible gases from the condenser during start up and normal operation without modifications. The review concluded that the main condenser evacuation system will continue to maintain its ability to control and provide monitoring for releases of radioactive materials to the environment following implementation of the SPU. Refer to Section 2.10.1, Occupational and Public Radiation Doses, for discussion of the adequacy of the plant regarding radioactive monitoring and control of releases of radioactive material including continued compliance with the requirements of GDC-60, -64, and 10 CFR 50, Appendix I. Therefore, the proposed SPU is acceptable with respect to the main condenser evacuation system.

2.5.3.3 Turbine Gland Sealing System

2.5.3.3.1 Regulatory Evaluation

The turbine gland sealing system is provided to control the release of radioactive steam in the turbine to the environment and prevent air in-leakage to the condenser. The Comanche Peak Nuclear Power Plant (CPNPP) reviewed changes to the turbine gland sealing system with respect to factors that may affect gaseous radioactive material handling (e.g., source of sealing steam, system interfaces, and potential leakage paths).

The acceptance criteria for the turbine gland sealing system are based on:

- General Design Criterion (GDC)-60, insofar as it requires that the plant design include means to control the release of radioactive effluents.
- GDC-64, insofar as it requires that means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences and postulated accidents.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

The adequacy of the CPNPP turbine gland sealing system design is assessed relative to conformance with the following:

- GDC-60, described in FSAR Section 3.1.6.1, Criterion 60 – Control of Releases of Radioactive Materials to the Environment, which states waste handling systems are incorporated in the facility design for processing and/or retention of radioactive wastes for normal operation and anticipated operational occurrences. Controls and monitoring are provided to ensure that releases during normal operation do not exceed a few percent of the limits of 10 CFR Part 20 and yield offsite doses within the numerical guides for design objectives and limiting conditions of operation set forth in 10 CFR Part 50, Appendix I.

  FSAR Section 9.4 describes the primary plant ventilation system and non-engineered safety features (ESF) exhaust units which satisfy GDC-60.

  FSAR Chapter 11 describes the radioactive waste processing systems' design criteria, holdup capacities, and estimated releases of radioactive effluents to the environment. Compliance with 10 CFR Part 50, Appendix I, is described in Appendix 11A.

- GDC-64, described in FSAR Section 3.1.6.5, Criterion 64 – Monitoring Radioactivity Releases. The Containment Building atmosphere is monitored continuously during normal and transient operations, using particulate, iodine, and gaseous monitors. Under post-accident conditions, samples of the containment atmosphere provide data of existing airborne radioactive effluent discharge paths and in the plant environs are continuously monitored during normal and accident conditions by the process radiation monitoring system described in FSAR Section 11.5. Provisions for monitoring the plant areas for radioactivity are included in the area radiation monitoring system described in Section 12.3.4. In addition to the installed detectors, periodic surveys are conducted using portable equipment as discussed in FSAR Section 12.5.

The system is designed as non-nuclear-safety-related in accordance with American National Standards Institute (ANSI) N18.2 and is classified as non-seismic Category I. The system is designed with provisions to monitor and control releases of gaseous radioactive material in accordance with GDCs-60 and -64 of 10 CFR Part 50.

2.5.3.3.2 Technical Evaluation

2.5.3.3.2.1 Introduction

The function of the turbine gland sealing steam system is to prevent leakage of steam to atmosphere and air in-leakage to the main condenser through the high-pressure and low-pressure turbine gland steam seals and the high-pressure and low-pressure stop and control valves. During startup, when the unit is under vacuum, steam is supplied from the seal steam header to the high-pressure and low-pressure shaft seals, and to the low-pressure stop and control valve stems to assure a slight positive pressure of approximately 4 inches H_2O, thereby preventing air from leaking into the turbine and affecting vacuum in the condenser. As the load increases and the high-pressure turbine internal pressure increases, the direction of the steam leakage reverses, and flows from the high-pressure turbine exhaust outward along the shaft. A small amount of steam leaks from the supply zone to suction zone, which is routed to the seal steam condenser. The remainder of the leakage steam exits from the high-pressure gland case to the steam supply header. The seal steam header system also receives bushing/stem leakages from the high-pressure and low-pressure stop and control valves. When the amount of steam leakoff from the high-pressure turbine seals and valve stems is sufficient to supply the low-pressure turbine seals, the header pressure will start to rise, and the supply valve will be gradually closed by the controller. As load is further increased, the supply valve will eventually be fully closed, and the leakoff control valve will gradually open to dump any excess steam in the header system to the low-pressure heater.

Air, noncondensable gases, and some water vapor are ejected by the main turbine gland steam condenser exhauster. The water drains to the atmospheric drain tank through a loop seal, and the air and noncondensable gases are vented to the primary plant ventilation exhaust system, which consists of 16 exhaust filtration units. Furthermore, the exhaust is monitored for radiation prior to release to the atmosphere.

2.5.3.3.2.2 Description of Analyses and Evaluations

The turbine gland sealing system was evaluated to ensure that the system design will continue to control the release of radioactive steam in the turbine to the environment. It will also continue to provide sufficient sealing steam to the high-pressure and low-pressure turbine glands from plant startup through full-power operation at the SPU conditions. The evaluation determined whether changes are required to the existing design of the system and its components in order to meet their design functions during SPU conditions and whether such changes affect the system's ability to control radioactive releases.

2.5.3.3.2.3 Results

The uprate does not change the startup functions of the seal steam system. The primary effect of the uprate is at full-load conditions, due to the increase in the high-pressure exhaust and hot reheat pressures. The uprate represents a change in high-pressure turbine exhaust pressure at 100-percent load, as follows:

On Unit 1, high-pressure exhaust pressure increases from the baseline pressure of 152.44 psia to 159.71 psia at the new 100-percent operating point, an increase of 7.27 psi.

On Unit 2, high-pressure exhaust pressure increases from the baseline pressure of 152.53 psia to 161.06 psia at the new 100-percent operating point, an increase of 8.53 psi.

Due to this change in pressure, the leakage across the high-pressure turbine inner gland is expected to increase by approximately 4.7 percent. Based on the header pressure being maintained at 4 inches of water, the sealing steam supply flow to the low-pressure glands, and the leakoff flows from all glands to the seal steam condenser should not change. Valve stem leakage from the high-pressure stop and control valves should remain approximately the same under SPU conditions, since the main steam inlet has not changed physically. Low-pressure stop and control valve steam leakages should increase in proportion to the increase in hot reheat pressure. Based on the flows for the plant, the total expected leakoff flow would be expected to increase by approximately 5.8 percent due to the uprate.

The leakoff valve is sized sufficiently to maintain the system pressure at the required pressure, with the valve approximately 80-percent open. At the same leakoff header pressure, assuming all other conditions remain approximately the same, the existing leakoff valve would be capable of passing approximately 27-percent additional flow. On that basis, the existing system should be capable of handling the additional 5.8-percent flow associated with the uprate.

In the event of primary-to-secondary leakage in the steam generator, there will be very low releases of radioactivity when the main turbine gland steam condenser is being bypassed.

The amount of noncondensable gases released by the condensed steam is negligible and there is no need for an activity monitor at the gland seal condenser exhauster. However, air and noncondensable gases are exhausted to the primary plant exhaust plenum, which is monitored and consists of a series of filtration units.

The evaluation of the turbine gland sealing system at SPU conditions demonstrates CPNPP Units 1 and 2 will continue to meet the current licensing basis with respect to the requirements of GDC-60, insofar as it requires that that plant design include means to control the release of radioactive effluents. This design capability remains unchanged by the SPU. The handling, control, and release of radioactive materials are in compliance with 10 CFR 50, Appendix I, as described in the Offsite Dose Calculation Manual.

The evaluation of the turbine gland sealing system at SPU conditions demonstrates that CPNPP Units 1 and 2 will continue to meet the current licensing basis with respect to the requirements of GDC-64, insofar as it requires that a means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and postulated accidents. This design capability remains unchanged by the SPU. Radioactivity levels contained in the effluent discharge paths in the environs are continually monitored during normal and accident conditions by the station radiation monitoring system and by the radiation protection program for CPNPP Units 1 and 2. Refer to LR subsection 2.10.1, Occupational and Public Radiation Doses.

2.5.3.3.3 Conclusions

Luminant Power concludes that the turbine gland sealing system will continue to maintain its ability to control and provide monitoring for releases of radioactive steam to the environment consistent with the current licensing basis with respect to the requirements of GDC-60 and GDC-64. Therefore, Luminant Power finds the proposed SPU acceptable with respect to the turbine gland sealing system.

2.5.4 Component Cooling and Decay Heat Removal

2.5.4.1 Spent Fuel Pool Cooling and Cleanup System

2.5.4.1.1 Regulatory Evaluation

The spent fuel pools provide wet storage of spent fuel assemblies. The safety function of the spent fuel pool cooling and cleanup (SF) system is to cool the spent fuel assemblies and keep the spent fuel assemblies covered with water during all storage conditions. The Comanche Peak Nuclear Power Plant (CPNPP) review focused on the effects of the proposed stretch power uprate (SPU) on the capability of the system to provide adequate cooling to the spent fuel during all operating and accident conditions.

The acceptance criteria for the spent fuel pool cooling and cleanup system are based on:

- General Design Criterion (GDC) -5, insofar as it requires that structures, systems, and components (SSCs) important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions.
- GDC-44, insofar as it requires that a system with the capability to transfer heat loads from safety-related structures, systems, and components to a heat sink under both normal operating and accident conditions be provided.
- GDC-61, insofar as it requires that fuel storage systems be designed with residual heat removal capability reflecting the importance to safety of decay heat removal, and measures to prevent a significant loss of fuel storage coolant inventory under accident conditions.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the adequacy of the CPNPP SF system design is assessed relative to conformance with the following:

- GDC-5, described in FSAR Section 3.1.1.5, General Design Criteria 5, Sharing of Structures, Systems, and Components, which states that the sharing of systems and components does not impair system capability to perform safety functions, e.g., the orderly shutdown and cooldown of one unit in the event of an accident in the other.
- GDC-44 is described in FSAR Section 3.1.4.15, General Design Criteria 44 – Cooling Water. GDC-44 addresses provision of a system to transfer heat from SSCs important

to safety to an ultimate heat sink (UHS). The system safety function shall be to transfer the combined heat load of these SSCs under normal operating conditions and accident conditions. The CPNPP includes redundant component cooling and service water design features to transfer heat to the UHS. The UHS has the capability to ensure either the simultaneous shutdown and cooldown of both units or the shutdown and cooldown of one unit simultaneously with the dissipation of post-accident heat from the other unit. Note that the SF system is not specifically addressed; however, the system does provide the heat removal design functions considered under this GDC.

- GDC-61 is described in FSAR Section 3.1.6.2, General Design Criteria 61 – Fuel Storage and Handling and Radioactivity Control. GDC-61 addresses the fuel storage and systems which contain radioactivity and shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed to include the following elements:

  1. The capability for periodic inspection and testing of components important to safety.
  2. Provisions for containment.
  3. Provisions for decay heat removal.
  4. The capability to prevent reduction in fuel storage coolant inventory under accident conditions.
  5. The capability and capacity to remove corrosion products, radioactive materials, and impurities from the pool water and reducing occupational exposures to radiation.

  FSAR Section 3.1.6.2 states that the SF system, which contains radioactivity, is designed as follows to ensure adequate safety under normal and postulated accident conditions:

  1. Components are designed and located such that appropriate periodic inspection and testing may be performed.
  2. All areas of the plant are designed with suitable shielding for radiation protection based on anticipated radiation dose rates and occupancy.
  3. Individual components that contain significant radioactivity are located in confined areas which are adequately ventilated through appropriate filtering systems.

4. The SF system provides cooling to remove residual decay heat from the fuel stored in the spent fuel pools and is designed with redundancy and testability to ensure continued heat removal. A purification loop is provided to remove fission product activity.

5. The spent fuel pools are designed so that no postulated accident can cause significant loss of coolant inventory.

6. The primary plant ventilation system is designed to filter the exhaust air from the fuel storage and handling area, radioactive waste and other ventilation systems which may contain radioactivity.

The SF system is described in FSAR Section 9.1.3, Spent Fuel Pool Cooling and Cleanup System. The CPNPP licensing bases regarding the SF system design features, operating modes, cooling capabilities, pool temperatures and failure modes are described in this FSAR section.

Other FSAR sections that address the design features related to the SF system include:

- FSAR Section 9.2.2, Component Cooling Water System, which describes the cooling water provided to the spent fuel pool heat exchangers.

- FSAR Section 9.4.2, Spent Fuel Pool Area Ventilation, which describes the ventilation system design and airborne activity control features provided for the spent fuel pool area.

2.5.4.1.2 Technical Evaluation

2.5.4.1.2.1 Introduction

As stated in FSAR Section 9.1.3, the system is designed to remove decay heat from fuel assemblies stored in the spent fuel pools. The system also maintains the clarity and purity of water in the spent fuel pools, the transfer canal, the wet cask pit, the refueling water storage tanks (RWST), and the refueling cavities.

The SF system consists of two cooling loops, two purification loops, and one surface skimmer loop. Each cooling loop includes a pump, heat exchanger, and associated piping, valves, and instrumentation. In the event of a failure of the spent fuel pool cooling pump or heat exchanger, the other loop ensures the continuity of effective cooling.

The heat removal criteria of the SF system are that the system should be capable of maintaining the temperature in each spent fuel pool less than or equal to 150°F during normal refueling operations with two loops, less than or equal to 200°F during normal refueling operations with a single active failure, and less than or equal to 212°F for emergency core offload situations. Normal refueling operations are conducted approximately every 18 months for each unit and the pool temperature is based on decay heat generation from a normal full core discharge (193 fuel

assemblies). The outages are assumed to begin and end during normal refueling periods (September 15th through May). The number of fuel assemblies expected to be permanently discharged for a given refueling outage is 96 fuel assemblies.

In each spent fuel pool, temperature elements and high temperature alarms are incorporated to alert operators to unexpected conditions in the fuel pools.

Description of Analyses and Evaluations

The SF system and components were evaluated to ensure they are capable of performing their intended functions at SPU conditions. The evaluations were performed for an analyzed core thermal power of 3,612 MWt. It is noted that no changes to system flows, pressures or temperatures are required due to SPU. The evaluations determined whether the existing design parameters of the spent fuel pool cooling system and components meet the SPU conditions for the following design aspects:

- Design pressure/temperature of piping and components
- Flow velocities
- Cooling capacity – Maximum design condition
- Cooling capacity – Maximum summer design
- Cooling capacity – Emergency core off-load
- Loss of cooling
- Concrete wall temperature
- Purification subsystem

Other evaluations related to the spent fuel pool cooling system and components are addressed in the following Licensing Report (LR) sections:

- Piping/component supports – Section 2.2.2.2, BOP (All Non-Class 1)

- Protection of the spent fuel pool cooling system from internally generated missiles – Section 2.5.1.2, Missile Protection

- Component cooling water for spent fuel pool heat exchangers – Section 2.5.4.3, Component Cooling Water

- Protection against dynamic effects of missiles, pipe whip and discharging fluids – Section 2.2.1, Pipe Rupture Locations and Associated Dynamic Effects and Section 2.5.1.3, Pipe Failures

- Fuel pool ventilation and control of airborne radioactivity – Section 2.7.4, Spent Fuel Pool Area Ventilation System

- Fuel pool criticality, fuel movement and storage; evaluation of the storage racks – Section 2.8.6.2, Spent Fuel Storage

2.5.4.1.2.2 Results

The evaluation of the spent fuel pool cooling system capabilities at SPU conditions demonstrates that the CPNPP will continue to meet the current licensing basis with respect to GDC-5, described in FSAR Section 3.1.1.5. The sharing of the spent fuel pool cooling system components does not impair the system capability to perform safety functions.

The evaluation of the spent fuel pool cooling system capabilities at SPU conditions demonstrates that the CPNPP will continue to meet the current licensing basis with respect to the requirements of GDC-44, described in FSAR Section 3.1.4.15. Although the spent fuel pool cooling system is not specifically addressed by this GDC, the system does provide for heat removal from the fuel pool and transfers the heat ultimately to the environment. The spent fuel pool cooling system provides this capability under both normal operating and accident conditions.

The evaluation of the spent fuel pool cooling system at SPU conditions demonstrates that the CPNPP will continue to meet the current licensing basis with respect to the requirements of GDC-61, insofar as it requires that systems that contain radioactivity be designed with appropriate periodic inspection and testing features, with suitable shielding for radiation protection, with appropriate containment, confinement, and filtering systems, with appropriate residual heat removal capability and with design features to prevent significant reduction in fuel storage coolant inventory under accident conditions. These design capabilities remain unchanged by the SPU. Refer to LR subsection 2.7.4, Spent Fuel Pool Area Ventilation, for additional details.

System/Component Design Parameters

No modifications to the spent fuel pool cooling system are required for SPU.

The maximum operating conditions at SPU do not change, including the maximum spent fuel pool temperature during cooling system operation of 150°F. Therefore, the existing design pressure and temperature of the system components; heat exchangers, pumps, valves, demineralizers, strainers, and filters are acceptable at SPU.

The current spent fuel pool cooling flow rate provides for acceptable heat removal in the spent fuel pool heat exchangers. Therefore, no changes are required to the fuel pool cooling pumps/motors and the system piping velocities are unchanged at SPU conditions.

Cooling Capacity – Maximum Design Condition

Normal refueling occurs approximately every 18 months and is based upon removal of 193 reactor core fuel assemblies. A minimum of four months between back to back outage starts is assumed. Assuming the core offload begins at 100 hours after shutdown and proceeds at a rate of 4 assemblies per hour, with the offload achieved at 150 hours after shutdown, the total decay heat load in the spent fuel pools is 57.9 MBtu/hr. With a service water temperature of 94°F, two loop operation, and aligning the cooling system so that pump 1 cools pool 1 and

pump 2 cools both pools, the system is capable of maintaining the spent fuel pools at or below 150°F.

The case of abnormal operation with one loop due to single active failure, the spent fuel pool cooling system is capable of maintaining the spent fuels pools at or below 200°F. For emergency core offload, the SF system is capable of maintaining the spent fuel pools at or below 212°F.

Cooling Capacity – Maximum Summer Design

The maximum summer design condition bounds the maximum normal heat loads which occur during normal power operation of both units. The SSI temperature is assumed to be the maximum allowable Technical Specification Limit (102°F). The total decay heat load in the spent fuel pools for this condition is 29.36 MBtu/hr. With these decay heat loads and two train operation, the system is capable of maintaining spent fuel pools temperature at or below 150.0°F. In the case of single train operation, the spent fuel pool cooling system is capable of maintaining the spent fuel pools at or below 200°F.

Cooling Capacity – Abnormal Maximum Design Condition

The abnormal maximum design condition bounds the abnormal heat load from an emergency core offload from either unit immediately after back to back refuelings. The total decay heat load in the spent fuel pools for this condition is approximately 67 MBtu/hr. The spent fuel pool cooling system is capable of maintaining the temperature of the spent fuel pools below 200°F.

Loss of Cooling

The spent fuel pool cooling system was evaluated for a loss of cooling. Assuming the maximum heat load in a single spent fuel pool is less than 52 MBtu/hr, the time to heat up the spent fuel pool from 150° to 212°F is greater than three hours. At this heat load, if cooling were not restored, a boil-off rate of approximately 110 gpm would exist, which is within the make-up capacity for the pool. Make-up water sources include the Refueling Water Storage Tank and the local fire hose stations which each have adequate supply capacity to accommodate these boil off rates.

Concrete Temperature

Since the existing spent fuel pool temperature requirements are being maintained for the uprate power level, there is no impact to the pool liner or concrete temperature as a result of the SPU/spent fuel pool (SFP) temperature.

Purification Subsystem

The SPU has no impact on the hydraulic portions of the purification subsystem. The current purification flow rate is adequate for SPU conditions. No equipment changes in the purification loop are required to support the uprate. The purification subsystem may experience a slight

increase in the frequency of demineralizer resin replacement due to higher levels of fission products in the pool. However, any significant increase in fission product inventory in the primary coolant system due to the SPU will be mitigated by RCS cleanup systems prior to transmission to the spent fuel pool.

2.5.4.1.3 Conclusions

Luminant Power has reviewed the assessment of the spent fuel pool cooling and cleanup system and concludes that the assessment has adequately accounted for the effects of the proposed SPU on the spent fuel pool cooling function of the system. Based on this review, Luminant Power concludes that the spent fuel pool cooling and cleanup system will continue to provide sufficient cooling capability to cool the spent fuel pool following implementation of the proposed SPU and will continue to meet the CPNPP current licensing basis with respect to the requirements of GDCs-5, -44, and -61. Therefore, the proposed SPU is acceptable with respect to the spent fuel pool cooling and cleanup system.

2.5.4.2 Station Service Water System

2.5.4.2.1 Regulatory Evaluation

The service water (SW) system provides essential cooling to safety-related equipment and also provides cooling to non-safety-related auxiliary components that are used for normal plant operation. The Comanche Peak Nuclear Power Plant (CPNPP) review covered the characteristics of the SW system components with respect to their functional performance as affected by adverse operational (that is, water hammer) conditions, abnormal operational conditions, and accident conditions (such as a loss-of-coolant accident (LOCA) with loss-of-offsite power). The review focused on the additional heat load that would result from the proposed stretch power uprate (SPU).

The acceptance criteria for the service water system are based on:

- General Design Criterion (GDC) -4, insofar as it requires that structures, systems, and components (SSCs) important to safety be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, including flow instabilities and loads (such as water hammer), maintenance, testing, and postulated accidents.

- GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions.

- GDC-44, insofar as it requires that a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions be provided.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

The adequacy of the CPNPP SW system design is assessed relative to conformance with the following:

- GDC 4, described in FSAR Section 3.1.1.4, Criterion 4 - Environmental and Dynamic Effects Design Bases, which states that SSCs important to safety are designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs. These SSCs are appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that can result from equipment failures and from events and conditions outside the nuclear power unit.

  Environmental conditions are described in FSAR Section 3.11, Environmental Design of Mechanical and Electrical Equipment.

- GDC-5, described in FSAR Section 3.1.1.5, Sharing of Structures, Systems, and Components. The sharing of systems or components does not impair the SW system's capability to perform safety functions, such as the orderly shutdown and cooldown of one unit in the event of an accident in the other unit.

- GDC-44, described in FSAR Section 3.1.4.15, Criterion 44 – Cooling Water, states that the cooling water system for safety-related functions consists of the SW system and the component cooling water (CC) system. The CPNPP includes redundant component cooling and SW design features to transfer heat to the ultimate heat sink. The ultimate heat sink has the capability to ensure either the simultaneous shutdown and cooldown of both units or the shutdown and cooldown of one unit simultaneously with the dissipation of post-accident heat from the other unit.

  Both the CC system and the SW system have two flow loops with redundant pumps, heat exchangers, and piping arrangements. The system is designed to meet the required safety function so that no single failure impairs cooling of essential equipment.

  Both systems are operable from either the offsite power system or the onsite diesel generators.

Additional service water system details are provided in FSAR sections:

- 9.2.1, Station Service Water System
- 9.2.2, Component Cooling Water System
- 6.2.4, Containment Isolation System

- 9.2.5, Ultimate Heat Sink
- 10.4.9, Auxiliary Feedwater System

2.5.4.2.2 Technical Evaluation

2.5.4.2.2.1 Introduction

The SW system is described in FSAR Section 9.2.1. The SW system removes heat from the CC system heat exchangers and from the emergency diesel generators, and supplies cooling water to the safety injection, centrifugal charging pump lube oil coolers, and the containment spray pump bearing oil coolers. In conjunction with the CC system, the SW system supplies cooling water to meet the plant cooling requirements during normal operation, shutdown, and during or after a postulated LOCA of either unit. The SW system also provides backup cooling water to the auxiliary feedwater system.

The SW system consists of two separate, independent, full-capacity safety-related trains. Cross-connections between the trains add operational flexibility to the SW system. The safety-related trains are redundant in that the components supplied by one train are sufficient to perform the minimum required safety functions.

2.5.4.2.2.2 Description of Analyses and Evaluation

The SW system and components were evaluated to ensure they are capable of performing their intended functions at SPU conditions. The evaluations compared the existing design parameters of the system/components with the SPU conditions for the following design aspects:

- SW flow and heat removal requirements
- Design pressure/temperature of piping and components
- Fouling in heat exchangers cooled by service water (Nuclear Regulatory Commission (NRC) Generic Letter 89-13)
- Loss of Cooling Water (NRC Generic Letter 91-13)

Other evaluations of the SW system and components are addressed in the following Licensing Report sections:

- Piping/component supports – Licensing Report (LR) subsection 2.2.2.2, BOP (Non-Class 1)
- Safety-related valve and pump testing and valve closure, including containment isolation requirements – LR subsection 2.2.4, Safety-Related Valves and Pumps
- SW system support of reactor cooldown requirements – LR subsection 2.8.4.4, Residual Heat Removal System

2.5.4.2.2.3 Results

Service Water Flow and Heat Removal from Cooled Components

The SPU evaluation has determined that the existing SW flow rates are sufficient to provide the required plant cooling. Therefore, the current service water pump capacities are acceptable for SPU conditions. The existing service water operating pressures are also unaffected since no physical changes to the SW system are required. The SW system will continue to operate at the current flow velocities and pressures.

The SW system is required to provide the necessary cooling water to the CC heat exchangers during all required plant operating conditions. The most limiting condition occurs during cooldown of the RCS when the CC system must meet the cooldown heat load in addition to heat loads from other plant equipment required to operate. During cooldown, the heat load to the CC heat exchangers are primarily affected by increased reactor decay heat at the SPU power level from the residual heat removal heat exchangers and added loads from the spent fuel heat exchangers. The heat loads from the CC heat exchangers to the SW system are controlled by limitations on the maximum allowable CC heat exchanger outlet temperatures, that is, during normal, cooldown, and post-accident conditions. These limitations are unaffected by the SPU and effectively limit the heat load from CC to service water during the higher heat portions of each condition analyzed. As a result of the SPU, there is a slight increase in the SFP cooling heat load applied to the SW system via CC water and ultimately to the safe shutdown impoundment. The resultant higher SW temperatures out of the CC heat exchangers remain bounded by the current design temperature of the piping and valves.

The heat loads from the diesel generator jacket water cooler, the centrifugal charging pump lube oil cooler, the safety injection pump lube oil cooler, and the containment spray pump bearing cooler remain unchanged at the SPU operating condition.

NRC Generic Letter 89-13

A CCW heat exchanger fouling monitoring program is employed at CPNPP to ensure that operating margin in the heat exchanger fouling is maintained. SPU may slightly reduce this operating margin for normal plant conditions.

NRC Generic Letter 91-13

FSAR 9.2.1.5 states that in the event of a loss of all essential cooling water on an operating unit, a cross-connect from the other unit may be opened to provide backup cooling capability. One service water pump is capable of providing essential cooling to both units for this event by manually aligning and flow balancing. Therefore, there is no adverse safety implication of this backup function.

2.5.4.2.3 Conclusion

Luminant Power has reviewed the assessment related to the effects of the proposed SPU on the station SW system. It concludes that the assessment has adequately accounted for the increased heat loads on system performance that would result from the proposed SPU. The review also concludes that the station SW system will continue to be protected from the dynamic effects associated with flow instabilities and provide sufficient cooling for SSCs important to safety following implementation of the proposed SPU. Therefore, it has been determined that the SW system will continue to meet the requirements of GDCs, -4, -5, and -44. Based on the above, Luminant Power finds the proposed SPU acceptable with respect to the SW system.

2.5.4.3 Reactor Auxiliary Cooling Water System

2.5.4.3.1 Regulatory Evaluation

This review covered the component cooling water system that is required for safe shutdown during normal operations, anticipated operational occurrences, and mitigating the consequences of accident conditions. The system includes a closed-loop auxiliary cooling water system for reactor system components, reactor shutdown equipment, ventilation equipment, and components of the emergency core cooling system (ECCS). The review covered the capability of the component cooling (CC) water system to provide adequate cooling water to safety-related emergency core cooling system components and reactor auxiliary equipment for all planned operating conditions. Emphasis was placed on the CC system as operated in conjunction with safety-related components (such as emergency core cooling system equipment, ventilation equipment, and reactor shutdown equipment). The review focused on the additional heat load that would result from the proposed stretch power uprate (SPU).

The acceptance criteria for the CC system are based on:

- General Design Criterion (GDC)-4, insofar as it requires that safety-related structures, system, and components (SSCs) important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation including flow instabilities and attendant loads (that is, water hammer), maintenance, testing, and postulated accidents.

- GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions.

- GDC-44, insofar as it requires that a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions be provided.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the Comanche Peak Nuclear Power Plant (CPNPP) GDC used during the licensing of CPNPP Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to these GDCs is discussed in FSAR Sections 3.1.1 through 3.1.5. Specifically, the adequacy of the CPNPP CC system design relative to conformance with GDCs-4, -5, and -44 is discussed in FSAR Sections 3.1.1 and 3.1.4 as follows:

- GDC-4, described in FSAR Section 3.1.1.4, Criterion 4 – Environmental and Dynamic Effects Design Bases. As described in this FSAR section, CPNPP SSCs important to safety are designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents (LOCAs). These SSCs are appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharge fluids that can result from equipment failures and from events and conditions outside the nuclear power unit.

- GDC-5, described in FSAR Section 3.1.1.5, Criterion 5 – Sharing of Structures, Systems, and Components. As described in this FSAR section, for those CPNPP facilities which have shared systems or components (such as CC system) the sharing of these systems or components does not impair system capability to perform safety functions (such as the orderly shutdown and cooldown of one unit in the event of an accident in the other unit).

- GDC-44, described in FSAR Section 3.1.4.15, Criterion 44 – Cooling Water. As described in this FSAR section, the cooling water system for safety-related functions consists of both the CC system and the station service water (SW) system (which removes heat from the component cooling water heat exchangers). The CC system is a closed system designed to remove heat from the reactor coolant system (RCS), cool the letdown flow to the chemical and volume control system (CVCS), cool safety-feature heat loads, and dissipate rejected heat from various plant components. The ultimate heat sink used to dissipate rejected heat from the reactor facility during normal and emergency shutdown conditions is the safe shutdown impoundment (SSI). The CC system has two flow loops with redundant pumps, heat exchangers, and piping arrangements. The system is designed to meet the required safety function so that no single failure impairs cooling of essential equipment.

  The ultimate heat sink has the capability to ensure either simultaneous shutdown and cooldown of both units or the shutdown and cooldown of one unit simultaneously with dissipation of post-accident heat from the other unit. The CC system (as well as the service water (SW) system) is operable from either the offsite power system or the onsite diesel generators.

Other FSAR sections that address the design features and functions of the CC system include:

- FSAR Section 9.2.2.2.2, Equipment Design Basis, which describes the design basis for the major components of the CC system. Major components include CC pumps, CC heat exchangers, CC surge tank, and piping and valves (in conjunction with FSAR Section 9.2.2.2.2, Table 9.2-2 provides a more detailed equipment listing and description showing design, performance, and material data for these components).

- FSAR Section 9.2.2.3.1, Safety Implications Related to Sharing, which describes CC system with respect to system portions shared between Units 1 and 2.

- FSAR Section 9.2.2.3.2, Leakage Detection and Control, which describes the design features for the detection of leakage through reactor coolant pressure boundary (RCPB) into the CC system.

- FSAR Section 3.5.1.1, Internally Generated Missiles (Outside Containment), which describes the design basis for those systems, including the CC system (as identified by FSAR Table 3.5-1), that require protection from missiles generated outside containment but internal to the plant.

- FSAR Section 3.2.1.1, Seismic Category I, which describes the seismic classification applicable to the CC system in accordance with the seismic requirements of GDC 2 of Appendix A to 10 CFR Part 50 (that is, systems which must remain functional during the safe shutdown earthquake are designated Seismic Category I).

- FSAR Section 6.2.4, Containment Isolation System, which describes the design features provided for containment isolation in conjunction with the CC system penetrations (that is, GDCs-56 and -57).

- FSAR Section 9.2.1, Station Service Water System (specifically Sections 9.2.1.2.1 and 9.2.1.8), which address service water fouling in CC heat exchangers (Nuclear Regulator Commission (NRC) Generic Letter (GL) 89-13), and CC heat exchanger fouling monitoring program.

2.5.4.3.2 Technical Evaluation

2.5.4.3.2.1 Introduction

The CC system is described in FSAR Section 9.2.2. The CC system is a safety-related system designed to supply cooling water to components that are part of the RCS, ECCS, engineered safety features (ESF) systems, CVCS, spent fuel pool cooling and cleanup system, waste processing system ventilation system, and Instrument air system. The CC system removes heat from plant components during all phases of plant operation, including startup, power operation, shutdown, refueling, and the injection and recirculation phases following a LOCA. The CC system, after picking up rejected heat from the various other systems, transfers the heat to the SW system via the component cooling water heat exchangers. The SW system, in

turn, transfers the heat to the ultimate heat sink (that is, SSI). Maximum normal operating SSI water temperature is 102°F, which is the value used for the evaluation of CC system safety-related and non-safety-related design features.

One CC system is provided for each unit. Each system contains a Train A safeguards loop, a Train B safeguards loop, and a non-safeguards loop (common components can be supplied by either Unit 1 CC or Unit 2 CC or a combination of the two). The two systems operate independently but can be cross-connected if needed.

Each system consists of two 100-percent capacity CC pumps and heat exchangers, with one pump and one heat exchanger operating, and one pump and one heat exchanger on standby. Each system consists of two separate, redundant and independent full-capacity safeguards loops to service the engineered safeguards components and a non-redundant non-safeguards loop with American National Standards Institute (ANSI) Safety Class 3 and non-safety class portions.

During normal full-power operation, each system has one CC pump and one CC heat exchanger in operation. A second CC pump and corresponding heat exchanger is on standby and aligned to replace the pump and heat exchanger in service. If the CC pump in operation trips, the redundant CC pump automatically starts and is operative within 60 seconds. During normal shutdown (requiring use of residual heat removal (RHR) system), two CC pumps and two CC heat exchangers are normally aligned to all loops of the subsystem serving the unit being cooled down. Component cooling water is supplied to both RHR heat exchangers.

The CC system also acts as an intermediate barrier between reactor coolant and the service water systems to ensure that leakage of radioactive fluid is contained within the plant. CC piping to/from the reactor coolant pumps thermal barrier and the associated isolation valves are designed to the reactor coolant system pressure and temperature to provide appropriate pressure/temperature boundary in the event of a rupture of the barrier. Radiation monitoring is provided to detect radioactivity entering the system from any of the cooled components and the system design includes the ability to isolate any cooled component when necessary.

2.5.4.3.2.2 Description of Analyses and Evaluation

The CC water system and components were evaluated to ensure they are capable of performing their intended functions at SPU conditions. The evaluations compared the existing design parameters of the system/components with the SPU conditions for the following design aspects:

- CC water heat exchanger performance (flow rates, duty and temperatures) at the increased SPU heat loads during normal power operation, normal cooldown, and abnormal transient and accident conditions

- CC water system temperature limits

- Design pressure/temperature of piping and components versus the SPU operating pressures and temperatures
- CC water relief valve capacities
- Protection of isolated piping sections from heatup effects

Other evaluations related to component cooling water system design are addressed in the following Licensing Report (LR) sections:

- Piping/component supports – LR subsection 2.2.2.2, BOP – Piping, Components and Supports (Non-Class 1)
- Protection against dynamic effects, including GDC-4 requirements, of missiles, pipe whip, discharging fluids and flooding – LR subsection 2.2.1, Pipe Rupture Locations and Associated Dynamic Effects; and LR subsection 2.5.1.3, Pipe Failures
- Component cooling water instrumentation – LR subsection 2.4.1, Reactor Protection, Engineered Safety Feature Actuation, and Control Systems
- Environmental qualification – LR subsection 2.3.1, Environmental Qualification of Electrical Equipment
- Safety-related valve and pump testing and valve closure, including containment isolation requirements – LR subsection 2.2.4, Safety Related Valves and Pumps
- Protection against turbine missiles and internal missiles – LR subsection 2.5.1.2, Missile Protection

2.5.4.3.2.3 Results

The following subsections evaluate the specific CC water system and component licensing, design and performance capabilities while at SPU conditions.

General Design Criteria

The evaluation of the CC water system capabilities at SPU conditions demonstrates that the CPNPP will continue to meet the current licensing basis with respect to the requirements of GDC-4. The system is protected from the dynamic effects of pipe break as described in LR subsection 2.2.1, Pipe Rupture Locations and Associated Dynamic Effects; and LR subsection 2.5.1.3, Pipe Failures. Safety-related equipment is environmentally qualified for the worst case environments as discussed in LR subsection 2.3.1, Environmental Qualification of Electrical Equipment. As described in LR subsection 2.5.1.3, Pipe Failures, the flooding analysis has considered the effects of moderate energy failures and evaluated the worst case failure in each plant building/area. The CC water system is a moderate energy system and previously analyzed failure effects are not affected by SPU conditions since the component

CC system flow rate and pressure does not change at SPU and no physical changes are being made.

The evaluation of the CC water system capabilities at SPU conditions demonstrates that CPNPP will continue to meet the current licensing basis with respect to the requirements of GDC-5. The CC water system includes two redundant pumps, two redundant heat exchangers, two redundant safeguards loops and one non-safeguards loop. The interconnections between Unit 1 safeguards loop and Unit 2 safeguards loop are blocked by one locked-closed isolation valve (as clarified in Supplemental Safety Evaluation Report (SSER) 22, separation between units safeguards loops is maintained by a single locked-closed isolation valve, which meets the separation requirements of GDC-5). CC water for Control Room air conditioning and uninterruptible power supply (UPS) air conditioning may be provided by either unit since these are shared systems. No physical changes are being made to the CC water system and no new operating modes or system lineups are required as a result of the SPU. Therefore, the CC water system continues to meet design requirements with respect to sharing of system and components in accordance with CPNPP licensing basis and GDC-5.

The evaluation of the CC water system capabilities at SPU conditions demonstrates that the CPNPP will continue to meet the current licensing basis with respect to the requirements of GDC-44. The CC water system provides heat removal from the reactor and plant equipment and transfers the heat ultimately to the environment via the ultimate heat sink. The component cooling water system provides this capability under both normal operating and accident conditions and is capable of achieving this function considering a single failure. The implementation of SPU does not affect the capability of the system to perform this function as demonstrated by the system and component evaluation results described below and by the analysis results discussed in LR subsection 2.6.1, Primary Containment Functional Design; and LR subsection 2.8.4.4, Residual Heat Removal System using the SW system during the postulated cooldown and accident scenarios.

CC Water Heat Removal Capability

CC water is provided to plant equipment (that is, system heat exchangers) in the safeguards loops and the non-safeguards loop. These heat exchangers are capable of removing the required SPU heat loads based on current CC flow rates. The SPU evaluation of this equipment is performed in the respective system-related LR sections referenced above.

Since none of the cooled components require more cooling flow, the existing CC water and service water flow rates through the component cooling water heat exchangers are not changed by the SPU.

Thermal performance of the CC system under SPU conditions was analyzed at the maximum Technical Specification allowable SSI temperature of 102°F. The lowest operating CC temperature is bound by a SW temperature of 40°F.

Under SPU conditions, at normal plant full-power operation, the heat loads from the cooled components are not significantly different from current loads, except for spent fuel heat loads. As a result, the CC supply temperature limits based on current operation are not significantly affected.

The cooldown cases provide the greatest heat loads and are the limiting cases relative to CC heat exchanger performance. The maximum CC water heat load during normal cooldown occurs when the RHR system is first placed in service, four hours after reactor shutdown. During cooldown (1 train), with maximum cooldown heat load conditions present, the initial RHR heat load is throttled to limit CC outlet temperature to a maximum temperature of 122°F (resulting in a longer cooldown time). The CC flow requirement is unchanged for the SPU and, consistent with current operation, the CC supply temperature will continue to be limited to 122°F during this operating mode.

Since, for the SPU condition, the CC heat exchanger performance limiting cases (that is, cooldown cases) continue to be governed by the 122°F CC supply temperature, the CC heat exchanger design capability remains acceptable during SPU operation.

During accident conditions (that is, post-LOCA), the CC temperature is maintained at 135°F or below during containment spray recirculation with an SW temperature of 102°F. The maximum CC temperature of the non-accident unit is limited to 122°F by operator actions upon initiation of RHR (several hours after the initiating event). The preceding maximum temperatures of 135° and 122°F ensure that the system will perform satisfactorily in mitigating the event in the DBA unit concurrent with the orderly shutdown and cooldown of the non-accident unit.

The SPU analyses described in LR subsection 2.6.1, Primary Containment Functional Design, confirm that the CC water heat exchangers provide sufficient heat removal capability for mitigation of postulated accidents.

SPU Operating Conditions Versus Design Conditions of Piping and Components

The existing CC water piping to/from the reactor coolant pump thermal barrier is designed for the reactor coolant system pressure and temperature in consideration of potential failure of the thermal barrier. The RCS design conditions do not change due to the SPU. Therefore, the piping design pressure and temperature for this portion of the CC water system is acceptable for SPU operation.

The CC water system flow rate does not change at SPU conditions and no physical changes are being made to the system (that is, no piping modifications and no pump modifications). Therefore, the CC water system operating pressures are not affected by SPU operation with the result that the existing piping and component design pressures continue to bound operating pressures, and are acceptable for SPU operation.

Slight changes in heat loads for various operating modes cause the CC water outlet temperatures from cooled components to increase slightly.

The CC water system has been evaluated relative to the impact of the SPU on system temperature design. All system temperature design parameters bound maximum CC system temperature parameters under SPU operation.

In turn, since CC water system piping, equipment, and component temperature design bounds CC system temperature design, all CC water system piping, equipment, and components are acceptable for operation under SPU conditions.

Component Cooling Water Relief Valve Capacities

The CC water system relief valves either have no change or have small changes in temperatures and are bounded by the relief valve design conditions. Since the SPU pressure/temperature condition is below the system design pressure/temperature, no additional analysis is required to demonstrate their acceptability.

The postulated flow, pressure, and temperature conditions from a failure of the reactor coolant pump (RCP) thermal barrier do not change for SPU operation since there are no changes to the existing reactor coolant system design conditions and no changes are being made to the RCP thermal barrier. Therefore, the relief valves on the CC water piping at the reactor coolant pump thermal barrier are unaffected by SPU conditions.

NRC Generic Letters 89-13 and 96-06

The NRC issues applicable to CPNPP in GLs 89-13 and 96-06 are related to service water fouling in heat exchangers (GL 89-13) and to heatup and overpressurization of isolated portions of piping inside containment, and boiling/water hammer in service water cooling lines to the containment atmosphere recirculation coolers (GL 96-06).

The issue in NRC GL 96-06 related to the heatup/overpressurization of isolated CC water piping inside containment was evaluated in previous responses to the NRC. Initially three areas of concern were identified two of which were associated with the CC water system piping. One of the concerns was resolved by implementation of a design change. The other concern was determined not to be an issue and no modifications were required. The conclusions relative to these original responses are not affected by SPU conditions since there are no physical changes or operational changes required by the SPU that would affect the containment penetration piping or isolation valves.

The small decrease in the containment post-accident temperature (Refer to LR subsection 2.6.1) at SPU conditions results in a value less than the value used in the original analysis. Therefore, no additional lines from the CC water system that penetrate the containment are considered a potential concern; no new relief valves are required and the existing relief valves remain acceptable.

The issue in NRC GL 89-13 related to evaluation of safety-related heat exchangers using service water and whether they have the potential for fouling therefore causing degradation in performance, and the mandate that there exist a permanent plant test and inspection program

to accomplish and maintain this evaluation. Luminant Power committed to, and subsequently implemented, a program to perform periodic inspection and preventative maintenance of CC heat exchangers. The conclusions relative to these original responses are not affected by the SPU since the existing procedures and activities in place at CPNPP in support of GL 89-13 are unaffected and require no changes (that is, subsequent to SPU, CC heat exchangers will continue to be periodically inspected and maintained).

More specifically, with respect to fouling factor parameters, the CPNPP program for monitoring fouling for heat exchangers in the SW system implements quarterly testing of CC heat exchangers (adjusted based on fouling factor or tube plugging margin). The fouling factor obtained, based on analysis of the test results, is compared with a set of curves that provide acceptable fouling factors. As a result of the SPU, the overall fouling factor associated with the CC heat exchanger has sufficient margin to meet requirements of single train cooldown. The tube plugging margin remains acceptable based on the current tube plugging levels.

2.5.4.3.3 Conclusion

The system review has assessed the effects of the proposed SPU on the component cooling water system and has adequately accounted for the increased heat loads from the proposed SPU on system performance. The system review concluded that the CC water system will be protected from the dynamic effects associated with flow instabilities and provide sufficient cooling for SSCs important to safety during all operating modes following implementation of the proposed SPU. Therefore, it is determined that the component cooling water system will continue to meet the current licensing basis with respect to the requirements of GDCs-4, -5, and -44. Based on the above, Luminant Power concludes that the proposed SPU is acceptable with respect to the CPNPP CC water system.

2.5.4.4 Ultimate Heat Sink

2.5.4.4.1 Regulatory Evaluation

The ultimate heat sink (UHS) is the source of cooling water provided to dissipate reactor decay heat and essential cooling system heat loads after a normal reactor shutdown or a shutdown following an accident. The Comanche Peak Nuclear Power Plant (CPNPP) review focused on the impact of the proposed stretch power uprate (SPU) on the decay heat removal capability of the UHS. Additionally, the review included evaluation of the design basis UHS temperature limit to confirm that post-licensing data trends (such as air and water temperatures, humidity, wind speed, or water volume) do not establish more severe conditions than previously assumed.

The acceptance criteria for the ultimate heat sink are based on:

- General Design Criterion (GDC) -5, insofar as it requires that structures, systems, and components (SSCs) important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions

- GDC-44, insofar as it requires that a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions be provided.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

The adequacy of the CPNPP UHS design is assessed relative to conformance with the following:

- GDC-5 is described in FSAR Section 3.1.1.5, Criterion 5 - Sharing of Structures, Systems, and Components. The sharing of the UHS, specifically the safe shutdown impoundment (SSI), by Units 1 and 2 does not impair the system capability to perform its safety functions (that is, the simultaneous shutdown and cooling down of both units, or the shutdown and cooldown of one unit concurrent with the dissipation of post-design-basis-accident rejected heat from the other unit). The SSI has been sized for the simultaneous shutdown and cooling down of both units, which is the bounding case for CPNPP).

- GDC-44 is described in FSAR Section 3.1.4.15, Criterion 44 - Cooling Water. The cooling water system for safety-related functions consists of the station service water (SW) system and the component cooling water (CC) system. The CC system is a closed system and is designed to remove residual heat from the reactor coolant system (RCS), to cool the letdown flow to the chemical and volume control system (CS), to cool safety-feature heat loads, and to dissipate rejected heat from various plant components. The SW system removes heat from the CC heat exchangers and transfers the heat to the SSI.

  Both the CC system and the SW system have two flow loops with redundant pumps, heat exchangers, and piping arrangements. The system is designed to meet the required safety function so that no single failure impairs cooling of essential equipment.

  The UHS has the capability to ensure either the simultaneously shutdown and cooldown of both units or the shutdown and cooldown of one unit simultaneously with the dissipation of post accident heat from the other unit.

The UHS was designed to meet the requirements of Regulatory Guide 1.27 (Revision 2, January 1976), as described in the following FSAR Sections:

- FSAR Section 2.3.1.2.10, Ultimate Heat Sink, describes the meteorological data for the design of the SSI.

- FSAR Section 2.4.11.5, Plant Requirements, states that the volume of water in SSI is sufficient to provide cooling for a period of over 30 days without makeup to mitigate the effects of an accident in one unit, to permit the safe shutdown of the other unit, and to maintain them both in a safe shutdown condition.

- FSAR Section 2.4.11.6, Heat Sink Dependability Requirements, describes the design feature of the SSI and the minimum SSI water level which occurs in the event of accidental water loss from the Squaw Creek Reservoir (SCR). The SCR normally supplies makeup to the SSI to maintain the same water level in the two water bodies. The section also states that the minimum water level is adequate for operation of the SW pumps

Additional UHS details are provided in FSAR Sections:

- 1.2.2.10, Safe Shutdown Impoundment
- 9.2.1, Station Service Water System
- 9.2.5, Ultimate Heat Sink

2.5.4.4.2 Technical Evaluation

2.5.4.4.2.1 Introduction

The function of the UHS is to dissipate heat rejected from the SW system during post-accident and normal cooldown conditions.

The UHS of CPNPP is the SSI, which is designed to dissipate heat rejected from the SW system. The SSI is an enclosed body of water formed from a cove of the SCR and is retained by a seismic Category I dam. The dam is designed and constructed to withstand the most severe postulated natural phenomena. An equalization channel connects the SCR and the SSI and allows each body of water to adjust to a common level above the minimum water level.

The SSI is designed to contain a water supply sufficient to allow simultaneous safe shutdown and cooldown of both units or safe shutdown of both units with one unit in loss-of-coolant accident (LOCA) allowing for a 30-day minimum reactor decay heat removal without outside makeup.

2.5.4.4.2.2 Description of Analyses and Evaluation

The UHS was evaluated to ensure it is capable of performing its intended function of a reliable water supply and heat removal capacity for normal and accident conditions following SPU.

The water temperature in the SSI is predicted using a three-dimensional (3-D) hydrothermal model that was specifically developed for the SSI. The primary input to the model is the increased transient heat loads to the SSI at SPU operating conditions. Because the rate of heat input to the SSI is controlled by design and procedures, the increased heat load only extends the length of the design transients. Other input data remain essentially unchanged from the

previous analysis. These include service water intake and discharge flows, meteorological data, and the SSI geometry. No changes are required to be made to the UHS Technical Specification that the SW intake temperature should not exceed 102°F.

No changes were made to the models used to analyze the SSI for SPU conditions.

Meteorological data for the period 1992 through 2006 were examined to determine if there was a period in which temperature or evaporation exceeds the values for limiting time periods utilized in the pre-SPU analysis. The critical year for temperature (1990 from the pre-SPU analysis) remains unchanged after consideration of the more recent meteorological data. Similarly, examination of the 30-day running average evaporation rates indicates the 1992-2006 meteorological data does not contain a more limiting period than the period in 1980 utilized in the pre-SPU analysis.

Previous thermal analyses of the SSI were performed at the engineered safety features rating of 3,565 MWt. Thermal analyses of the SSI were performed at SPU conditions. The slight increase in calculated SPU heat loads results in a slight increase in SSI temperature for the various scenarios considered. The maximum calculated temperature of 115.7°F for LOCA increased to 115.9°F. Evaporation also increased slightly as a result of the increase in heat loads from the engineered safety features rating (3,565 MWt) to SPU. The final SSI level due to evaporation decreased by 1 inch for LOCA.

2.5.4.4.3 Conclusion

Analysis of SSI temperature response for LOCA at SPU conditions indicates a maximum temperature increase of 0.2°F over the previous analysis of the maximum predicted peak temperature. Similarly, the analysis of evaporation for SPU showed a small increase over the previous analysis. These are small changes are well within the capability of the SSW pumps for NPSH and the essential equipment cooled by the SSWS.

Luminant Power has reviewed the evaluation related to the effects of the proposed SPU on the UHS. Luminant Power concludes that the proposed SPU will not compromise the design-basis safety function of the UHS, and that the UHS will continue to meet the CPNPP current licensing basis with respect to the requirements of GDC-5 and GDC-44. Therefore, Luminant Power finds the proposed SPU acceptable with respect to the UHS.

2.5.4.5 Auxiliary Feedwater System

2.5.4.5.1 Regulatory Evaluation

In conjunction with a seismic Category I water source, the auxiliary feedwater system functions as an emergency system for the removal of heat from the primary system when the main feedwater system is not available. The auxiliary feedwater system is also used to provide decay heat removal necessary for withstanding or coping with a station blackout. The Luminant Power review of the proposed stretch power uprate (SPU) focused on the system's continued ability to provide sufficient emergency feedwater flow at the expected conditions (such as steam generator pressure) to ensure adequate cooling with the increased decay heat.

The acceptance criteria for the auxiliary feedwater system are based on:

- General Design Criterion (GDC)-4, insofar as it requires that structures, systems, and components (SSCs) important to safety be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures.

- GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions.

- GDC-19, insofar as it requires that equipment at appropriate locations outside the Control Room be provided with (1) the capability for prompt hot shutdown of the reactor, and (2) a potential capability for subsequent cold shutdown of the reactor.

- GDC-34, insofar as it requires that a residual heat removal system be provided to transfer fission product decay heat and other residual heat from the reactor core, and that suitable isolation be provided to assure that the system safety function can be accomplished, assuming a single failure.

- GDC-44, insofar as it requires that a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions be provided, and that suitable isolation be provided to assure that the system safety function can be accomplished, assuming a single failure.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

The adequacy of the CPNPP auxiliary feedwater design is assessed relative to conformance with the following:

- GDC-4, described in FSAR Section 3.1.1.4, Criterion 4 – Environmental and Dynamic Effects Design Bases, which states that SSCs important to safety are designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents (LOCAs). These SSCs are appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that can result from equipment failures and from events and conditions outside the nuclear power unit.

  Environmental conditions are described in FSAR Section 3.11, Environmental Design of Mechanical and Electrical Equipment.

- GDC-5, described in FSAR Section 3.1.1.5, General Design Criteria 5, Sharing of Structures, Systems, and Components. Units 1 and 2 do not share any portion of the auxiliary feedwater system.

- GDC-19, described in FSAR Section 3.1.2.10, Criterion 19 – Control Room. Safe occupancy of the Control Room during abnormal conditions is provided for in the design. The Control Room is in a seismic Category I structure. Adequate shielding is provided for the Control Room in the event of a design basis accident (DBA), as detailed in FSAR Sections 12.1 and 15.6. The Control Room air conditioning system features redundant equipment, and radiation and smoke detectors, with appropriate alarms and interlocks. Provisions are made for Control Room air to be recirculated through high-efficiency particulate air (HEPA) and charcoal filters.

  In the unlikely event that access to the Control Room is restricted, the hot shutdown panel, local control stations, or manual operation of critical components can be used to trigger hot shutdown from outside the Control Room for an extended period.

- GDC-34, described in FSAR Section 3.1.4.5, Criterion 34 – Residual Heat Removal. The residual heat removal (RHR) system, in conjunction with the steam and power conversion system, is designed to transfer the fission production decay heat and other residual heat from the reactor core within acceptable limits. The auxiliary feedwater system provides backup for the steam and power conversion system.

  The crossover from the steam and power conversion system to the RHR system occurs at approximately 350°F and 425 psig.

  Suitable redundancy at temperatures below approximately 350°F is accomplished with the two RHR pumps (located in separate compartments with means available for draining and monitoring of leakage), the two heat exchangers, and the associated piping, cabling, and electric power sources. The RHR system is capable of operating on

either onsite or offsite electrical power systems. Details of the residual heat removal system are in FSAR Section 5.4.7.

- GDC-44 is described in FSAR Section 3.1.4.15, General Design Criteria 44 – Cooling Water. The cooling water system for safety-related functions consists of the station service water (SW) system and the component cooling (CC) water system. Both the SW and CC have two flow loops with redundant pumps, heat exchangers, and piping arrangements. The system is designed to meet the required safety function so that no single failure impairs cooling of essential equipment. The ultimate heat sink has the capability to ensure either the simultaneous shutdown and cooldown of both units or the shutdown and cooldown of one unit simultaneously with the dissipation of post-accident heat from the other unit. Both systems are operable from either the offsite power system or the onsite diesel generators.

Other FSAR sections that address the design features and functions of the auxiliary feedwater system include:

- FSAR Section 6.2.1.4.4 Auxiliary Feedwater System Design, which describes the operation and failure modes of the auxiliary feedwater system during postulated steam line breaks.
- FSAR Section 6.2.4, Containment Isolation System, which describes the containment isolation features to isolate the containment boundaries in the auxiliary feedwater system post accident.
- FSAR Section 10.4.9, Auxiliary Feedwater Systems, which describes the Auxiliary Feedwater design basis, system operation and instrumentation.
- The auxiliary feedwater system is also credited in the mitigation of the transient and accident analyses described in the FSAR Chapter 15 analyses.

2.5.4.5.2 Technical Evaluation

2.5.4.5.2.1 Introduction

The auxiliary feedwater system is described in FSAR Section 10.4.9. The auxiliary feedwater system supplies feedwater to the secondary side of the steam generators at times when the feedwater system is unavailable, and during all unit heatups and cooldowns. It provides a cooling source in the event of a small-break LOCA. Additionally, the system is used in the event of a main steam line break, feedwater line break, loss of power, or low-low steam generator level conditions. Under loss-of-offsite-power conditions, the auxiliary feedwater system maintains the plant at hot standby conditions. Under loss of all AC power (station blackout), the turbine driven auxiliary feedwater pump remains capable of auto or manual start.

The auxiliary feedwater system consists of three separate pump trains. Two of the trains consist of individual and separate branches utilizing motor-driven auxiliary feedwater pumps (MDAFPs). One train consists of an individual and separate branch utilizing one turbine-driven auxiliary

feedwater pump (TDAFP). Normally each MDAFP supplies two steam generators, but the alignment can be altered to allow either MDAFP to supply any two or all four steam generators. The TDAFP normally supplies feedwater to all four steam generators. Each pump supplies the steam generators through a normally open, motor-operated, discharge valve. The preferred source of water for the auxiliary feedwater system is the condensate storage tank (CST). A long-term source of water is available through a cross-tie with the SW.

2.5.4.5.2.2 Description of Analyses and Evaluations

The auxiliary feedwater system and associated components were evaluated to ensure they are capable of performing their intended functions at the SPU conditions. The evaluations compared the existing design parameters of the systems/components with the SPU conditions in conjunction with the following design aspects:

- Required flow rates/pump capabilities
- Design versus operating pressure/temperature of piping and components
- Containment isolation capabilities
- Water supplies/sources
- Pump design and performance

The primary impact of the SPU on the auxiliary feedwater system is the increased core thermal power and the resulting higher heat removal requirements during abnormal, transient, and accident conditions.

Other evaluations of the auxiliary feedwater system, piping and components are addressed in the following Licensing Report (LR) sections:

- Piping/component supports and water hammer effects – LR subsection 2.2.2.2, BOP (All Non-Class 1)

- Steam supply to the turbine driven auxiliary feedwater pump – LR subsection 2.5.5.1, Main Steam

- Operation of the auxiliary feedwater system during postulated abnormal and accident scenarios is discussed in LR subsection 2.8.5, Accident and Transient Analyses

- Operation of the auxiliary feedwater system during station blackout is discussed in LR subsection 2.3.5, Station Blackout

- Protection against dynamic effects, including GDC-4 requirements, of missiles, pipe whip and discharging fluids – LR subsection 2.2.1, Pipe Rupture Locations and Associated Dynamic Effects, and LR subsection 2.5.1.3, Pipe Failures

- Auxiliary feedwater instrumentation – LR subsection 2.4.1, Reactor Protection, Engineered Safety Feature Actuation, and Control Systems

- Environmental qualification of pumps and valves – LR subsection 2.3.1, Environmental Qualification
- Safety related valve and pump testing and valve closure, including containment isolation requirements – LR subsection 2.2.4, Safety-Related Valves and Pumps
- Protection against turbine missiles and internal missiles is discussed in LR subsection 2.5.1.2, Missile Protection

2.5.4.5.2.3 Results

The higher heat removal requirements during abnormal cooldown and accident conditions do not change the maximum operating conditions of auxiliary feedwater system piping and components with implementation of the proposed SPU. As such, auxiliary feedwater system piping and components are acceptable for SPU operation. Since there are no flow changes with the SPU the likelihood of fluid flow instabilities is not increased with the SPU.

SPU accident and transient analyses demonstrate the existing auxiliary feedwater system arrangement/performance remain bounding for SPU operations in terms of providing adequate flow and pressure to mitigate the consequences of the design basis events/accidents, and support normal cooldown and safety grade cold shutdown operations.

The feedline break analysis requires a minimum flow of 430 gpm to the 3 intact steam generators prior to isolation of the broken feedwater line. This is an increase from the current requirement of 400 gpm. This impacts the minimum operating point of the TDAFP; and the in-service testing (IST) surveillance acceptance criteria will be revised. There is adequate pump margin and head capacity to accommodate this change.

In the event of a loss-of-offsite power (LOOP), sufficient useable inventory must be available in each unit's CST to bring the unit from full power to hot standby conditions, maintain the unit at hot standby for 4 hours and then cool down to RHR system entry conditions in 5 hours. The minimum required useable volume at the uprated condition is 241,000 gallons. The current Technical Specification useable volume is greater than the 241,000 gallons needed. Therefore, no change to the Technical Specification 3.7.6 is required for uprate. The implementation of the SPU will not impact the CST.

Implementation of the SPU will not impact the CST and the SW to function as the primary and secondary source, respectively, for the auxiliary feedwater system.

The design pressures/temperatures of the existing auxiliary feedwater pumps are acceptable for operation under SPU conditions. There are no physical changes to the existing auxiliary feedwater pumps or to the service water pumps that supply water at pressure to the pumps' suctions. Therefore, there is no change to worst case operating pressure under SPU conditions and the existing manufacturer's design pressure ratings for the auxiliary feedwater pumps remain bounding.

Since there are no physical changes to the CST, auxiliary feedwater piping, auxiliary feedwater pumps, and the worst-case flow rate has not changed, the net positive suction head (NPSH) required by the auxiliary feedwater pumps remains the same pre- and post-SPU.

The brake horsepower required by the auxiliary feedwater pumps is unaffected by SPU conditions since there are no changes to the existing pump design and the maximum operating flow rate. Therefore, there is no additional horsepower demand on the existing motors of the auxiliary feedwater pumps due to SPU operation.

The steam to the turbine driver of the TDAFP is supplied by the main steam system. The evaluation of the steam supply capability concluded that sufficient steam can be supplied at SPU conditions to meet the brake horsepower requirements of the TDAFP. See LR subsection 2.5.5.1, Main Steam, for details.

The safety-related valves in the auxiliary feedwater system were reviewed to confirm that the SPU conditions of flow and differential pressure do not affect valve operation. Since, under SPU operation, there are no increases in the worst-case auxiliary feedwater system flow rates and pressures, and no hardware changes to the auxiliary feedwater pumps, the safety related valves are not impacted and are acceptable for operation under SPU conditions. See LR subsection 2.2.4, Safety-Related Valves and Pumps.

2.5.4.5.3 Conclusions

The evaluation has adequately accounted for the effects of the increase in decay heat and other changes in plant conditions on the ability of the auxiliary feedwater system to supply adequate water to the steam generators to ensure adequate cooling of the core. The auxiliary feedwater system will continue to meet its design functions following implementation of the proposed SPU. Luminant Power concludes that the auxiliary feedwater system will meet the current licensing basis with respect to the requirements of GDC-4, -5, -19, -34, and -44. Therefore, the proposed SPU is acceptable with respect to the auxiliary feedwater system.

2.5.5 Balance-of-Plant Systems

2.5.5.1 Main Steam System

2.5.5.1.1 Regulatory Evaluation

The main steam system transports high quality wet steam from the nuclear steam supply system (NSSS) to the power conversion system and various auxiliaries. The main steam system has both safety-related and non-safety-related functions. The Luminant Power review focused on the effects of the proposed stretch power uprate (SPU) on the system's capability to transport steam to the power conversion system, provide heat sink capacity, supply steam to drive safety system pumps, and withstand adverse dynamic loads (such as steam hammer resulting from rapid valve closure and relief valve fluid discharge loads). The Nuclear Regulatory Commission's (NRC's) acceptance criteria for the main steam system are based on:

- General Design Criterion (GDC) -4, insofar as it requires that safety-related structures, systems, and components (SSCs), be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures.
- GDC-5, insofar as it requires that safety-related SSCs not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions.
- GDC-34, insofar as it requires that a residual heat removal be provided to transfer fission product decay heat and other residual heat from the reactor core.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC for the plant SSCs important to safety comply with Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. Specifically, the adequacy of the Comanche Peak Nuclear Power Plant (CPNPP) main steam system design relative to conformance to:

- GDC-4 is described in FSAR Section 3.1.1.4, General Design Criteria 4 – Environmental and Missile Design Bases. Conformance relating to the requirements of GDC-4 is also described in the following:
  - Environmental Design of Mechanical and Electrical Equipment (FSAR Section 3.11N)
  - Protection Against the Dynamic Effects Associated with the Postulated Rupture of Piping (FSAR Section 3.6N & 3.6B)
    - Pipe breaks inside containment
    - Pipe breaks outside containment
  - Missile Protection (FSAR Section 3.5)

- GDC-5 is described in FSAR Section 3.1.1.5, General Design Criteria 5 – Sharing of Structures, Systems, and Components, which states that CPNPP Unit 1 and Unit 2 have no shared SSCs in the main steam system.

- GDC-34 is described in FSAR Section 3.1.4.5, General Design Criteria 34 – Residual Heat Removal. FSAR Section 10.3 provides details as to how the main steam system is used to transfer fission product decay heat from the reactor core.

- The main steam header dynamic load factor analysis is discussed in FSAR Section 3.9B.3.4.2, Nuclear Piping (American Society of Mechanical Engineers (ASME) Classes 2 and 3).

2.5.5.1.2 Technical Evaluation

2.5.5.1.2.1 Introduction

The main steam system is described in FSAR Section 10.3. The system provides heat removal from the reactor coolant system during normal, accident, and post-accident conditions. During off-normal conditions, the system provides emergency heat removal from the reactor coolant system using secondary heat removal capability.

The main steam system is designed to transport high quality saturated steam that was produced in the steam generators to the high-pressure turbine, as well as other steam-driven components and auxiliary systems. The portions of the main steam system from the steam generators up to and including the main steam isolation valves, the atmospheric relief valves, and the main steam safety valves are designed as safety related. The main steam supply to the auxiliary feedwater pump turbine and turbine exhaust piping is also safety-related. The main steam system also provides a flow path for steam from the steam generators to the steam dump system (The steam dump system is discussed in Licensing Report (LR) subsection 2.5.5.3, Turbine Bypass).

The reheat steam system is considered part of the main steam system for the CPNPP. The reheat system includes the cold reheat piping, the moisture separator reheaters (MSRs), hot reheat piping, valves, and the piping associated with this equipment. The reheat system delivers steam from the high-pressure turbine exhaust through the MSRs and then to the low-pressure turbine inlets. In the MSRs, moisture contained in the steam is removed; steam is dried and then super-heated (using main steam) and sent to the low-pressure turbines.

The main steam system is described in the FSAR Section 10.3, Main Steam Supply System. Additional main steam system details are provided in FSAR Section 5.4.9, Main Steam Line and Feedwater Piping; FSAR Section 6.2.4, Containment Isolation System; FSAR Section 7.4.1.1.2, Steam Generator Atmospheric Relief Valves and Main Steam Safety Valves; and FSAR Section 10.1, Summary Description.

2.5.5.1.2.2 Description of Analyses and Evaluations

The main steam and reheat steam systems and components were evaluated to ensure they are capable of performing their intended functions at SPU conditions. The evaluations compared the existing design parameters of the systems/components with the SPU conditions for the following design aspects:

- Design pressure/temperature of piping and components versus SPU conditions
- SPU flow thermal conditions and flow velocities
- Main steam line pressure drops at SPU conditions
- Vibration due to increased flow
- Capacities, closure times, and set pressures for the main steam isolation valves, main steam safety valves, and atmospheric relief valves
- Steam turbine evaluation
- MSR evaluation
- Main steam supply capacity to the auxiliary feedwater pump turbine and to other auxiliary loads
- Main steam drain system capacity

Other evaluations of main steam and reheat steam systems and components are addressed in the following LR sections:

- Erosion/corrosion issues – LR subsection 2.1.8, Flow-Accelerated Corrosion
- Piping/component supports – LR subsection 2.2.2.2, BOP (All Non-Class 1)
- Pipe Rupture Locations and Associated Dynamic Effects; and LR subsection 2.5.1.3, Pipe Failures
- Environmental qualification of the main steam isolation valves actuators – LR subsection 2.3.1, Environmental Qualification
- Main steam isolation valve testing and closure requirements – LR subsection 2.2.4, Safety-Related Valves and Pumps
- Protection against turbine missiles and internal missiles in accordance with GDC-4 is discussed in LR subsection 2.5.1.2.1, Missile Protection

- Main steam and reheat steam instrumentation – LR subsection 2.4.1, Reactor Protection, Engineered Safety Features Actuation, and Control Systems

- Turbine control/overspeed protection – LR subsection 2.5.1.2.2, Turbine Generator

2.5.5.1.2.3 Results

System Operating Conditions

Heat balance models were developed to determine the steam cycle parameters while operating at the increased NSSS power level. Heat balances were developed for the current power level based on actual plant operating data and at the SPU NSSS power level of 3,628 MWt. The SPU heat balances specify the required main steam flow rates and the high-pressure turbine inlet throttle pressures for both units. The steam pressures at turbine throttle valve inlet are 983 psia for Unit 1 and 959 psia for Unit 2. The process parameters are derived from turbine manufacturer's uprate hardware data.

Based on the CPNPP heat balances, the main steam conditions at the steam generator outlet and the reheat steam conditions to/from the MSRs are listed in Table 2.5.5.1-1.

Piping Evaluations

Design Pressure/Temperature

The main steam system design pressure/temperature of 1,185 psig (1,200 psia) and 600°F bound the full power SPU operating conditions of approximately 1,000 psia and 545°F. Additionally, the highest normal operating pressures and temperatures occur at no-load conditions. These conditions are not affected by the SPU.

The reheat crossover/crossunder piping were specified with the design conditions of 245 psig/545°F for the crossunder piping (high-pressure turbine exhaust to the moisture separator reheaters) and 218 psig/850°F for the crossover piping (MSRs to the low-pressure turbine inlet). These original design conditions bound the SPU operating pressure of 152 psig and temperature of 523°F. The MSR safety valves existing set pressures and capacities are adequate for SPU operation and provides overpressure protection for reheat steam system (including piping).

The design pressure and temperature for the steam supply piping to the main feedwater pump turbine bound SPU conditions.

Flow Velocities

Flow velocities through the main steam piping from the steam generators to the steam turbine control and stop valves were calculated at current and SPU conditions. The flow velocities increased approximately 5 percent primarily due to the increased flow required by the SPU power level. At SPU NSSS power, the steam velocity is 109 feet per second (fps) which is

below the industry design guideline. SPU steam velocities in the main steam supply piping to MSRs increased approximately 4 percent with the highest velocity at 141 fps, again below the industry design guideline.

Flow velocities in the other parts of the main steam and reheat steam system are within the industry design guideline and acceptable for SPU operation.

Vibration

The increased steam flow velocity through piping and components has the potential to increase vibrations. Accordingly, during power ascension, piping and components will be monitored to identify line vibration anomalies. These vibration monitoring activities are discussed in LR Section 2.12, Power Ascension and Testing Plan.

Component Design Evaluations

Design Pressure/Temperature

As described above under Piping Evaluations, Design Pressure/Temperature, the main steam system design pressure and temperature are not affected by SPU operation. The design conditions of the main steam components – main steam isolation valves, atmospheric relief valves, and main steam safety valves – were reviewed and, in all cases, were equal or above the main steam system design conditions of 1,185 psig and 600°F, and bound the SPU operation conditions.

The MSR's shells have a design pressure and temperature of 195 psig/600°F, which envelope the SPU operating pressures. The MSR tubes contain main steam for heating and have a design pressure and temperature of 1,250 psig/600°F which envelope the main steam design conditions of 1,185 psig/600°F and bound the SPU operation conditions

Main Steam Safety Valves Capacities and Setpoints

The setpoints of the main steam safety valves are determined based on the design pressure of the steam generators and main steam system (1,185 psig) and is a requirement of the ASME Boiler and Pressure Vessel (B&PV) Code. Since the design pressures of the steam generators and main steam system have not changed for the SPU, there is no need to revise the setpoints of the safety valves.

The main steam safety valves must have sufficient capacity so that the 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 has been applied so that the valves should be sized to relieve 105 percent of the maximum calculated steam flow at an accumulation pressure not exceeding 110 percent of the main steam system design pressure.

CPNPP has 20 main steam safety valves for each unit with a total rated capacity of 18.19×10^6 lb/hr, which provides about 111.7 percent of the maximum SPU full-load steam flow of 16.29×10^6 lb/hr. Therefore, the capacity of the installed main steam safety valves meets the Westinghouse sizing criterion.

The original design requirements for the main steam safety valves included a maximum flow limit per valve of 970,000 lb/hr at 1,185 psig. Since the actual capacity of any single main steam safety valve is less than the maximum flow limit per valve, the maximum capacity criteria is satisfied.

Atmospheric Relief Valves Capacities and Setpoints

The main steam atmospheric relief valves are located upstream of the main steam isolation valve and the main steam safety valves. They are automatically controlled by steam line pressure during normal plant operations. The atmospheric relief valves open and exhaust to atmosphere whenever the main steam line pressure exceeds a predetermined setpoint to minimize safety valve lifting during steam pressure transients. As the main steam line pressure decreases, the atmospheric relief valves close and reset at a pressure below the opening pressure. The atmospheric relief valve set pressure for normal operations is between zero-load steam pressure and the setpoint of the lowest-set main steam safety valves. Since there are no changes to these pressures for the NSSS, there is no need to change the atmospheric relief valve setpoints.

The primary function of the atmospheric relief valves is to provide a means for decay heat removal and plant cool down by discharging steam to the atmosphere when the condenser, the condenser circulating water pumps, or steam dump to the condenser is not available. Under such circumstances, the atmospheric relief valves in conjunction with the auxiliary feedwater system permit the plant to be cooled down from the pressure setpoint of the lowest-set main steam safety valves to the point where the residual heat removal system can be placed in service.

The main steam atmospheric relief valves are sized to have a minimum capacity of 62,150 lb/hr at 100 psia to support reactor cool down to residual heat removal system operating conditions in 5 hours; the capacity is sufficient to achieve a cooldown rate of 50°F/hr. This design basis is limiting with respect to sizing of the atmospheric relief valves and bounds the capacity required for a steam generator tube rupture event.

Since the CPNPP instrument air system is not a nuclear safety-related system, a nuclear safety-related air accumulator is provided for operating each atmospheric relief valve when instrument air is unavailable. The SPU has no effects on the existing accumulator.

An evaluation has been performed for CPNPP atmospheric relief valves at SPU conditions. The evaluation concluded that the original design bases, in terms of plant cooldown capability, can still be achieved with the installed atmospheric relief valves capacities under the SPU conditions.

Main Steam Isolation Valves

The main steam isolation valves are designed for a maximum pressure of 1,200 psia (1,185 psig), which is equal to the main steam system design pressure. The SPU operating pressures are the same for Unit 1 and slightly lower than current operating pressures for Unit 2; the design pressure of 1,185 psig is not affected by the SPU. Therefore, the valve design is unaffected.

The impact of the higher main steam flow rates through these valves during SPU operation was evaluated to confirm that the valves will close within the required time period of 5 seconds. The closure time is not affected since the valve and operator design is based on the flow rate due to the worst-case break flow that the valve experiences. The SPU does not affect the pipe break flows since the factors that affect maximum possible break flow, such as break size, location, steam generator pressure, and exit nozzle characteristics, and so forth are not affected by SPU. Refer to LR subsection 2.2.4, Safety-Related Valves and Pumps.

Steam Turbine Generator

The CPNPP 1,800 rpm turbine train is composed of three elements: one double-flow high-pressure turbine and two double-flow low-pressure turbines with a last row rotating blade approximately 46 inches long. The cycle includes an MSR between the outlet of the high-pressure turbine and the inlets of the low-pressure turbines. The MSR unit consists of a four pass, single moisture separator stage and one steam reheat single stage supplied by main steam. Refer to FSAR Sections 10.2.2.1 and 10.2.2.7.7, which discuss the steam turbine and turbine stop and control valves.

An evaluation was performed for CPNPP Unit 1 of the critical areas of the turbine generator to identify any components that may require modification in order to accommodate the new steam flow, pressure, temperature, and power output at the uprate plant conditions. The study results bound CPNPP Units 1 and 2 steam turbines.

The evaluation determined the following:

- For the high-pressure turbine, the uprate thermal conditions are within the current design limits. However, the existing high-pressure turbine rotor and inner casing steam path will be replaced to accommodate the increase in steam flow from the SPU to achieve the MWe generation increase. The high-pressure turbine outage casing will not be replaced.

- The low-pressure turbine uprate steam conditions are within the current design conditions and are acceptable for SPU operation with no modification.

- Main steam stop valves and control valves (high-pressure turbine) pressures and temperatures as well as the velocities as a results of the uprating, are all within the current design limits of the unit, and therefore acceptable to operate at the SPU conditions.

- The low-pressure turbine inlet stop valves and control valves pressures, temperatures, and the velocities as a results of the uprating, are all below the current design limits, and therefore acceptable to operate at the SPU conditions.

- The expected turbine overspeed is proportional to the increase in power output. The overspeed setpoint is currently 110 percent of rated speed. The overspeed design limit is 120 percent of rated speed. The current expected overshoot is 8.4 percent; approximately 118.4 percent of design limit. After the uprating, the expected overshoot will increase to 9.0 percent; approximately 119 percent of design limit. Based on this value, there is no required change to the overspeed setpoint.

- The total drain requirements during startup will remain unaffected by the SPU. Based on the conditions per the SPU heat balance, the amount of water that is required to be removed by the cold reheat piping (crossunder piping) drain line is slightly decreased compared to the baseline conditions per current operation heat balance. This is due to the slight decrease in the moisture content of the high-pressure exhaust steam, and the effect of the higher pressure on the empirical moisture separation factor used in the calculation of the moisture separation rate.

For Missile Protection, see LR subsection 2.5.1.2.

<u>Moisture Separator Reheaters Safety Valves</u>

The MSR safety valves are provided as overpressure protection for the MSR vessels and for the piping from the high-pressure turbine exhaust through the MSRs and to the low-pressure turbine inlet valves. The relieving capacity and setpoints of the valves must be sufficient to pass the SPU operating hot reheat steam flow exiting the MSRs and maintain the ASME pressure limits on the MSRs and connected piping. It had been concluded that the existing MSR safety valves have sufficient relieving capacity and the safety valve set pressures are adequate and do not need to be changed under SPU operation.

<u>Moisture Separator Reheaters</u>

The moisture separator reheaters were evaluated to determine the impact of the increased steam flow and pressure during SPU operation on tube vibration, thermal performance, moisture removal capability, internal shell drain, and erosion/corrosion effects.

The evaluation concluded:

- The reheater tube bundles will endure the SPU flow-induced vibration; there is no vibration problems associated with SPU operation.

- The reheater tube bundles will perform sufficiently to meet the thermal requirements of SPU operation at 100-percent power generation.

- The moisture removal capability of the moisture separator and the internal drain system is sufficient for the SPU moisture loading. MSR drain piping is addressed in LR subsection 2.5.5.4, Condensate and Feedwater.

- It is not expected that significant flow-acceleration corrosion (FAC) will occur in the MSR at SPU conditions. See LR subsection 2.1.8, Flow-Accelerated Corrosion.

Therefore, the MSRs are acceptable for operation at the SPU without modifications.

Main Steam Supply to Auxiliary Feedwater Pump Turbine

The auxiliary feedwater pump turbine steam supply piping and exhaust piping were determined to be acceptable for SPU conditions. The pressure ratings remain bounding. The required steam supply flow rate to the pump turbine is not affected by the SPU since the design power of the auxiliary feedwater pump turbine bounds the power required to supply the maximum SPU auxiliary feedwater flow rate. See LR subsection 2.5.4.5, Auxiliary Feedwater System.

Main Steam Piping Drain Capacity

The main steam piping is provided with drains to collect water from steam condensed in the piping. The drains were originally designed with sufficient capacity based on startup conditions where hot steam is introduced into cold piping. Startup steam conditions, such as flow, quality, temperature and pressure, are not affected by the SPU. Also, since there are no changes to the main steam piping arrangement, the existing drain system remains acceptable.

2.5.5.1.3 Conclusions

The operating pressure and temperature associated with the SPU conditions are bounded by the design of the main steam system piping and components. The main steam flow velocities at SPU conditions are below industry guidelines and remain acceptable. Pressure drops in main steam and reheat steam lines are acceptable as shown by the SPU heat balances, which confirm that the predicted thermal performance of the plant results in the expected SPU MWe output.

The main steam safety valves design conditions, capacity, and setpoints are adequate for SPU operation.

The steam generator atmospheric relief valves design conditions, capacity, and setpoints are adequate for SPU operation.

The main steam isolation valves and operators are acceptable for SPU flow conditions and will satisfactorily close in the required time period because the existing valve design parameters of flow rate and pressure bound the SPU conditions.

The MSRs of the reheat steam system will continue to provide sufficient moisture removal capability and adequate thermal performance during SPU operation and will operate without

excessive vibration or erosion/corrosion at the SPU flow rates and pressure/temperature conditions.

The MSR safety valves have adequate relieving capacity to provide overpressure protection for the MSR shell and connecting piping. The remainder of the reheat piping is acceptable since their design pressure/temperatures envelope the SPU conditions.

The evaluation concludes that the main steam system will maintain its ability to transport steam to the power conversion system, provide heat sink capacity and supply steam to steam-driven safety pumps. Main steam system steam hammer is addressed in LR subsection 2.2.2. The residual heat removal system is addressed in LR subsection 2.8.4.4. The evaluation of SPU implementation further concludes that the main steam system will continue to meet the CPNPP current licensing basis with respect to the requirements of GDCs-4, -5 and -34 as described in FSAR Section 3.1. The main steam piping, components, and supports (all Non-Class 1) to meet the requirements for GDC-4 is addressed in LR subsection 2.2.2.2. Therefore, the proposed SPU is acceptable with respect to the main steam system.

Table 2.5.5.1-1

System Operating Conditions

	Current Operating Condition		**SPU Operating Condition**	
	Unit 1	**Unit 2**	**Unit 1**	**Unit 2**
Main Steam – Steam Generator Outlet				
Flow Rate, lbm/hr	15,362,900	15,338,500	16,167,200	16,114,800
Pressure, psia	990.0[1]	982.0	1,000.0	976.0[1]
Temperature, °F	543.4	542.4	544.6	541.6
Reheat Steam – MSRs Inlet (Crossunder)				
Flow Rate, lbm/hr	11,345,900	11,420,100	11,904,600	12,005,300
Pressure, psia	158.5	157.9	166.1	166.4
Temperature, °F	362.8	362.5	366.6	366.7
Reheat Steam – MSRs Outlet (Crossover)				
Flow Rate, lbm/hr	9,698,400	9,723,600	10,165,700	10,263,900
Pressure, psia	151.8	151.9	159.0	160.
Temperature, °F	522.6	512.6	523.0	510.8

Note:

1. The operating pressure following the replacement of steam generators at Unit 1 is 1,000 psia, which has little impact on the current operating conditions identified in this table. The operating pressure at Unit 2, 976 psia, is based on a steam generator tube plugging level of 1.0%.

2.5.5.2 Main Condenser

2.5.5.2.1 Regulatory Evaluation

The main condenser system is designed to condense and deaerate the exhaust steam from the main turbine and provide a heat sink for the turbine bypass system. The Comanche Peak Nuclear Power Plant (CPNPP) review focused on the effects of the proposed stretch power uprate (SPU) on the steam bypass capability with respect to load rejection assumptions, and on the ability of the main condenser system to withstand the blowdown effects of steam from the turbine bypass system.

The acceptance criteria for the main condenser system are based on:

- General Design Criterion (GDC)-60, insofar as it requires that the plant design includes means to control the release of radioactive effluents.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

The adequacy of the CPNPP main condenser system design is assessed relative to conformance with the following:

- GDC-60, described in FSAR Section 3.1.6.1, Criterion 60 – Control of Releases of Radioactive Materials to the Environment, which states waste handling systems are incorporated in the facility design for processing and/or retention of radioactive wastes for normal operation and anticipated operation occurrences. Controls and monitoring are provided to ensure that releases during normal operation do not exceed a few percent of the limits of 10 CFR Part 20 and yield offsite doses within the numerical guides for design objectives and limiting conditions of operation set forth in 10 CFR Part 50, Appendix I.

  FSAR Section 9.4 describes the primary plant ventilation system and non-engineered safety features (ESF) exhaust units that satisfy GDC-60.

  FSAR Chapter 11 describes the radioactive waste processing systems' design criteria, holdup capacities, and estimated releases of radioactive effluents to the environment. Compliance with 10 CFR Part 50, Appendix I, is described in FSAR Appendix 11A.

2.5.5.2.2 Technical Evaluation

2.5.5.2.2.1 Introduction

There are two low-pressure (LP) turbines per unit. One LP turbine discharges steam into condenser A, and the other discharges steam into condenser B. Each LP turbine has two exhaust paths into its respective condenser shell. There are two tube bundles within each condenser shell, each tube bundle being located under one of the LP turbine exhaust paths. Thus, there are a total of 4 tube bundles per unit, 2 per condenser shell, and 1 per LP turbine discharge flow path. The original CPNPP condensers (copper tubes) were replaced with titanium tubes in 1987.

The steam spaces within condenser shells A and B are connected by a large diameter pipe so that there is pressure equalization between them.

The condensed steam within each condenser shell falls into a hot well at the bottom of each shell. There is one hot well that collects the condensate under both tube bundles in each condenser shell.

The hot well for condenser A and the hot well for condenser B collect the condensed water that provides suction to the two condensate pumps in each unit. The condensate within each hot well flows out of a large opening in the bottom of the hot well and almost immediately these two flow paths join together into one large diameter pipe that carries the mixed condensate from the two hot wells into the suction of the two operating condensate pumps.

2.5.5.2.2.2 Description of Analyses and Evaluations

The main condenser will experience higher steam flows due to the increase in LP turbine exhaust flow at the SPU power level during normal power operation. The evaluation determined the impact of the SPU conditions on condenser performance and integrity as follows:

- Determine the increased condenser duty and confirm the condenser's ability to reject heat to the circulating water system and maintain a low enough condenser backpressure for the turbine to meet its SPU MW output and performance requirements.

- Evaluate performance of the condenser based on general operating cases for: turbine operation, bypass operation, and combined bypass operation plus turbine operation.

- Evaluate the impact of higher condenser backpressure at the SPU conditions.

- Evaluate tube vibration at the SPU conditions.

- Evaluate condenser tube erosion at higher SPU conditions.

- Evaluate the impact of the increased steam flow on the plant design to control the release of radioactive effluents in accordance with GDC-60.

2.5.5.2.3 Results

The evaluation determined that the condenser satisfactorily removes the increased SPU heat loads, condenses the required steam flows and maintains an acceptable vacuum using circulating water at the current normal operating flow rate. Table 2.5.5.2-1 describes the key design parameters of the main condenser and compares its performance at current operating and SPU conditions.

Evaluation of the condenser performance shows that significant change is not expected in normal operation at the SPU conditions.

Evaluation of the condenser from the current operating value to the SPU operating value for backpressure effects was performed for the cooling water inlet temperature range from 60° to 105°F and found to be acceptable. Table 2.5.5.2-1 compares the condenser performance for 95° and 102°F cooling water inlet temperatures. The 95°F cooling water inlet temperature is the original design specification. The current bounding summer condition of 101°F cooling water inlet temperature is projected to become 102°F under SPU operating conditions. At an average of 82°F cooling water inlet temperature, the resulting condenser backpressure incremental value is 0.09 inch HgA.

Tube vibration can be caused by high steam velocities in the condenser shell. An evaluation of tube vibration was performed that took into account the lowest cooling water temperature of the last three years (66°F). In addition, no vibration concerns have been reported for the last 20 years. The evaluation concluded that normal operation at SPU conditions was acceptable from a vibration standpoint.

Currently, there are no erosion issues in the CPNPP condensers. An evaluation was performed based on an empirical correlation for the "erosion number," developed by Siemens to analyze condensers for erosion effects. Results indicate an insignificant difference in erosion number between current operating conditions and the SPU conditions.

The current turbine trip and alarm set points for condenser backpressure are not affected by the increased steam flow rates at SPU conditions for normal operation.

The design of the main condenser does not change following the implementation of the SPU. Therefore, the SPU does not impact the ability of CPNPP regarding the control of radioactive material in accordance with GDC-60. Monitoring of the air and non-condensibles leaving the condenser is accomplished by a radiation monitor in the condenser evacuation system, described in LR subsection 2.5.3.2, Main Condenser Evacuation System. The impact of the SPU on radiological effluent releases from the CPNPP, radiation monitoring setpoints and compliance with 10 CFR 50, Appendix I, are discussed in LR subsection 2.10.1, Occupational and Public Radiation Doses.

2.5.5.2.4 Conclusions

Luminant Power has reviewed the assessment of the effects of the proposed SPU on the main condenser system and concludes the assessment adequately accounted for the effects of changes in plant conditions on the design of the main condenser. Luminant Power concludes that the main condenser system will continue to maintain its ability to withstand the blowdown effects of the steam from the turbine bypass system and thereby, continue to meet the CPNPP current licensing basis with respect to the requirements of GDC-60 for preventing the consequences of failures in the system. Therefore, Luminant Power finds the proposed SPU is acceptable with respect to the main condenser.

Table 2.5.5.2-1

Main Condenser Performance Characteristics

	Condenser Design Specification		**Current Operating Value 100 % Power**		**SPU Operating Value 100% Power**	
Condenser Duty	7.759×10^9 Btu/hr		7.684×10^9 Btu/hr		8.056×10^9 Btu/hr	
Cooling Water Inlet Temperature	95°F	102°F	95°F	102°F	95°F	102°F
Cooling Water Temperature Rise	15.2°F	15.2°F	15.1°F	15.1°F	15.8°F	15.8°F
Condenser Backpressure	3.52 inch HgA	4.22 inch HgA	3.42 inch HgA	4.19 inch HgA	3.55 inch HgA	4.33 inch HgA

2.5.5.3 Turbine Bypass

2.5.5.3.1 Regulatory Evaluation

The turbine bypass system, which at the Comanche Peak Nuclear Power Plant (CPNPP) Units 1 and 2 is referred to as the steam dump system, is designed to discharge a portion of main steam flow directly to the main condenser system, bypassing the turbine. This steam bypass enables the plant to take step-load reductions up to the steam dump system capacity without the reactor or turbine tripping. The system is also used during startup and shutdown to control steam generator pressure.

Luminant Power's review focused on the effects that a SPU has on load rejection capability, analysis of postulated system piping failures, and on the consequences of inadvertent steam dump system operation. The acceptance criteria for the steam dump system are based on (1) General Design Criterion (GDC)-4, insofar as it requires that structures, systems, and components (SSCs) important to safety be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures; and (2) GDC-34, insofar as it requires that an a residual heat removal (RHR) system be provided to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits (SAFDLs) and the design conditions of the reactor coolant pressure boundary (RCPB) are not exceeded.

Current Licensing Basis

Criterion 4 – Environmental and Dynamic Effects Design Bases – "Structures, Systems, and Components Important to Safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These SSCs shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, the dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Nuclear Regulatory Commission (NRC) demonstrate that the probability of fluid system piping rupture is extremely low under basis conditions fluid system piping."

Discussion

The station's SSCs important to safety are designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accident (LOCA). Environmental conditions are described in Final Safety Analysis Report (FSAR) Section 3.11.

These SSCs are appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that can result from equipment failures and from events and conditions outside the nuclear power unit.

Details of the design, environmental testing, and construction of these systems, structures, and components are included in FSAR Chapters 3, 5, 6, 7, 8, 9, and 10. Evaluation of the performance of safety features is contained in FSAR Chapter 15.

Criterion 34 – Residual Heat Removal – "A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded."

Discussion

The RHR system, in conjunction with the steam and power conversion system, is designed to transfer the fission production decay heat and other residual heat from the reactor core within acceptable limits. The auxiliary feedwater system provides backup for the steam and power conversion system in this function. The auxiliary feedwater system is described in FSAR Section 10.4.9.

The crossover from the steam and power conversion system to the RHR system occurs at approximately 350°F and 425 psig.

Details of system design are in FSAR Section 5.4.7.

2.5.5.3.2 Technical Evaluation

The steam dump system was evaluated to ensure that the system was capable of performing its intended function for the range of NSSS design parameters (Licensing Report (LR) Section 1.1) approved for the SPU.

The steam dump system creates an artificial steam load by dumping steam from ahead of the turbine valves to the main condenser. The Westinghouse original sizing criterion conservatively 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 nuclear steam supply system (NSSS) to withstand an external load reduction of up to 50 percent of plant-rated electrical load without a reactor/turbine trip. To prevent a trip, this transient requires all NSSS control systems to be in automatic, including the rod control system, which accommodates 10 percent of the load reduction. The steam dump system prevents main steam safety valve (MSSV) lifting following a reactor trip from full power.

Each CPNPP unit is equipped with 12 condenser steam dump valves. Each valve is currently sized to have a flow capacity of about 845,292 lbm/hr at a valve inlet pressure of 930 psia. This valve capacity provides a total steam dump system capacity that exceeds the original Westinghouse sizing criterion.

The capacity of the steam dump system (as a percentage of full-load steam flow) decreases as full-load steam pressure decreases and full-load steam flow increases. However, NSSS operation within the proposed range of design parameters for SPU will not result in a reduced

steam dump capability relative to the original Westinghouse sizing criteria. An evaluation indicates steam dump capacity could be as low as 45.2 percent of rated steam flow (16.15×1^{06} lb/hr), or 7.303×10^6 lb/hr at a full-load steam pressure equal to 804 psia. At full-load steam pressures higher than 804 psia (T_{avg} = 574.2°F), steam dump capacity would increase. For example, at a full-load steam pressures of 1,005 psia (T_{avg} = 589.2°F), steam dump capacity would be 61.9 percent of rated flow (14.97×10^6 lb/hr), or 9.268×10^6 lb/hr.

As described in LR subsection 2.4.1, Reactor Protection, Safety Features Actuation, and Control Systems, no system hardware changes are required to ensure the plant will continue to satisfactorily respond to design basis loading and unloading described in LR subsection 2.4.2, Plant Operability.

The NSSS stability and operability analysis described in LR subsection 2.4.2.1, Plant Operability, provides an evaluation of the adequacy of the steam dump system in conjunction with the control system setpoints at SPU conditions. This section states that a 50-percent load rejection with steam dumps available to the condenser can be accommodated without resulting in either a reactor/turbine trip or steam generator safety valve actuation. The analysis results indicate that for the range of NSSS design parameters approved for SPU, a turbine trip from less than 50 percent of the uprate thermal power can be accommodated without challenging the pressurizer PORVs for full-load operating T_{avg} values greater than or equal to 580°F with the current steam dump control system loss of load controller setpoints. For reduced full-load operating T_{avg} values less than 580°F, the loss of load controller setpoints must be revised in order not to challenge the pressurizer PORVs. Furthermore, a step decrease equivalent to 50 percent of the uprated thermal power can be accommodated without a reactor trip occurring for full load T_{avg} values greater than or equal to 580°F with the current loss of load controller setpoints and for reduced full load operating T_{avg} values less than 580°F with revised loss of load controller setpoints.

Based on these analyses, the condenser steam dumps meet the requirements at SPU conditions as discussed above.

The condenser steam dump valves have NSSS requirements on time for opening and for modulating steam flow. To provide effective control of flow on large step-load reductions or plant trip, the steam dump valves are required to go from full closed to full open in 3 seconds at any pressure between 50 psi less than full-load pressure and steam design pressure. The dump valves are also required to modulate to control flow. For modulating steam dump flow, the positioning response may be slower with an allowed maximum full-stroke time of 20 seconds. These time response requirements are not affected by the SPU and must still be met.

The current plant safety analysis limits the capacity of the steam dump valves, as well as the steam generator atmospheric relief valves (ARVs) and MSSVs to a maximum flow limit per valve of 308 lb/sec at 1,185 psig. The capacity of any single steam dump valve, ARV or MSSV is not changed for the SPU. LR subsection 2.8.5.6.1, Inadvertent Opening of a Steam Generator ARV, Steam Dump Valve or Safety Valve, demonstrates that the results of a stuck-open steam generator ARV, steam dump valve, or safety valve are bounded by the large (hypothetical) steam line break results.

2.5.5.3.3 Conclusion

Luminant Power has reviewed the assessment of the effect of the proposed SPU on the steam dump system. Luminant Power concludes that the evaluation has adequately accounted for the effects of changes in plant conditions on the design of the system. It is also concluded that the steam dump system will continue to provide a means for shutting down the plant during normal operations. It is further concluded that steam dump system failures will not adversely affect essential systems or components. Based on this, Luminant Power concludes that the steam dump system will continue to meet GDCs-4 and -34 and finds the proposed SPU acceptable with respect to the steam dump system.

2.5.5.4 Condensate and Feedwater System

2.5.5.4.1 Regulatory Evaluation

The condensate and feedwater system provides feedwater at the appropriate temperature, pressure, and flow rate to the steam generators.

The Comanche Peak Nuclear Power Plant (CPNPP) review focused on the effects of the proposed SPU in previous analyses and considerations with respect to the capability of the condensate and feedwater system to supply adequate feedwater during plant operation and shutdown, and to isolate components, subsystems, and piping in order to preserve the system's safety function. The safety-related portion of the condensate and feedwater systems is the feedwater piping portion between the moment restraints upstream of the outermost Feedwater containment isolation valves and the steam generator feedwater nozzles. This portion is designed to seismic Category I requirements.

The Nuclear Regulatory Commission's (NRC's) acceptance criteria for the condensate and feedwater system are based on:

- General Design Criterion (GDC)-4, insofar as it requires that structures, systems, and components (SSCs) important to safety be designed to accommodate the effects of and be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, and that such SSCs be protected against dynamic effects.
- GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions.
- GDC-44, insofar as it requires that a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions be provided, and that suitable isolation be provided to ensure that the system safety function can be accomplished, assuming a single failure.

Current Licensing Basis

The adequacy of the CPNPP design relative to the general design criteria is discussed in Final Safety Analysis Report (FSAR) Section 3.1. Specifically, the adequacy of CPNPP condensate and feedwater system design relative to conformance to:

- GDC-4 is described in FSAR Section 3.1.1.4, Criterion 4 - Environmental and Dynamic Effects Design Bases. Conformance to the requirements of GDC-4 is described in the following:

  The station's SSCs important to safety are designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation,

maintenance, testing, and postulated accidents, including loss-of-coolant accident (LOCA). Environmental conditions are described in FSAR Section 3.11N.

These SSCs are appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that can result from equipment failures and from events and conditions outside the nuclear power unit. Conformance to the requirements of GDC-4 is described in the following:

- Environmental Design of Mechanical and Electrical Equipment (FSAR Section 3.11N)
- Protection Against the Dynamic Effects Associated with the Postulated Rupture of Piping (FSAR Section 3.6B) (Licensing Report (LR) subsection 2.5.1.3)
- Missile Protection (FSAR Section 3.5) (LR subsection 2.5.1.2)

Details of the design, environmental testing, and construction of these SSCs are included in FSAR Chapters 3, 5, 6, 7, 8, 9, and 10. Evaluation of the performance of safety features is contained in FSAR Chapter 15.

- GDC-5 is described in FSAR Section 3.1.1.5, Criterion 5 – Sharing of Structures, Systems, and Components. CPNPP is a dual unit installation. The FSAR identifies that the shared systems or components are tabulated in Section 1.2. Section 1.2 does not list the Condensate or Feedwater systems.

- GDC-44 is described in FSAR Section 3.1.4.15, Criterion 44, Cooling Water. GDC-44 addresses provision of a system to transfer heat from safety-related SSCs to an ultimate heat sink. The system safety function shall be to transfer the combined heat load of these SSCs under normal operating and accident conditions. Note that the condensate and feedwater systems are not specifically addressed. However, the feedwater system does have safety functions, which are considered as part of this GDC, by providing redundant flow paths for the auxiliary feedwater system flow to the steam generators for heat removal from the reactor coolant system and by providing the required safety related, redundant isolation functions of main feedwater during postulated steamline breaks. The replacement steam generator project for Unit 1 replaced the Model D-4 Steam Generators with Westinghouse Model Delta 76 Steam Generators. As a result, there are differences in the feedwater piping to the steam generators. Unit 1 has a single feedwater line to each Steam Generator. The auxiliary feedwater system provides its water to the steam generator via its own nozzle connection. The Unit 2 feedwater design with the Westinghouse model D-5 Steam Generators has a split flow configuration, which includes an 18-inch connection to the steam generator above the tubesheet and a 6-inch split flow connection above the tubes. For Unit 2, upper feedwater connections are used to deliver the auxiliary feedwater system to the steam generators.

Other FSAR sections that address the design features and functions of the condensate and feedwater systems include:

- FSAR Section 6.2.1.4, Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures, which describes the isolation requirements for the feedwater system during postulated pipe breaks.
- FSAR Section 6.2.4, Containment Isolation System, which describes the containment isolation features to isolate the feedwater system containment boundaries.
- FSAR Chapter 15 Accident Analysis, which describe the isolation of feedwater lines, containment isolation and auxiliary feedwater supply (Unit 2) via the main feedwater headers to the steam generators during the transient and accident analyses described in the FSAR.
- FSAR Section 10.4.7.2, Systems Description describes the feedwater control system that mitigates water hammers in feedwater piping.

2.5.5.4.2 Technical Evaluation

2.5.5.4.2.1 Introduction

The condensate and feedwater systems are described in FSAR Section 10.4.7. The condensate and feedwater systems are designed to provide feedwater to the steam generators during steady-state operation at the maximum guaranteed turbine load.

Safety-related components and piping within the feedwater system are used for containment isolation and feedwater isolation during accidents and transients as well as being the main feedwater flow paths to each steam generator during normal operation. In Unit 2 only, safety-related components and piping are also used for auxiliary feedwater addition. For Unit 1 the auxiliary feedwater does not utilize the Unit 1 feedwater system piping.

The pre SPU condensate and feedwater system design functions include:

- The condensate portion of the system is designed to supply approximately 9.85×10^6 lb/hr to the suction side of the main feedwater pumps during steady-state operation at maximum guaranteed turbine load.
- In addition, the condensate system can supply 96 percent of the full-load feedwater flow (14.54×10^6 lb/hr) to the main feedwater pumps during load-drop transients.
- The feedwater portion of the system is designed to supply the feedwater required for various loads at steady-state operation and to maintain this flow, as required, during the steam dump conditions following a large load reduction.

- The system is designed to maintain uniform feedwater flow to all steam generators under all conditions and to maintain proper steam generator water levels automatically during steady-state and transient conditions.

2.5.5.4.2.2 Description of Analyses and Evaluations

The condensate and feedwater systems and components were evaluated to ensure they are capable of performing their intended functions at SPU conditions. The evaluation considered the effects of the SPU on the following system/component design aspects:

- Design pressures/temperatures of piping, valves and components versus SPU operating pressures/temperatures

- Flow velocities

- Feedwater isolation valves closure within the required time period at SPU hydraulic conditions of flow and pressure drop

- Capacity and control capability of the feedwater flow control valves

- Feedwater heaters design parameters and operating characteristics

- Pump and pump supporting subsystems design capabilities, including net positive suction head (NPSH), flow, head, brake horsepower, minimum flow protection, and seal water supplies

- Process setpoints for protective functions, such as pump NPSH. The condensate and feedwater systems were evaluated by utilizing a hydraulic model of the system components and piping and the SPU heat balances. Physical plant data for the installed components and piping were utilized in the hydraulic model. Physical changes to condensate and feedwater components, valves and piping that resulted from the SPU evaluations were incorporated into the hydraulic model and verified as acceptable.

Current plant operating data were gathered and included in the current operating heat balances to reflect the present day performance of the existing components.

The SPU heat balances were used to establish the flow, temperatures and heat transfer requirements at the SPU power level.

Other evaluations of condensate and feedwater systems and components are addressed in the following LR sections:

- Effects of increased flow and velocity on erosion/corrosion concerns – LR subsection 2.1.8, Flow Accelerated Corrosion

- Piping, component and supports – LR subsection 2.2.2.2, Balance-of-Plant Piping (Non-Class 1)
- Protection against dynamic effects, including GDC-4 requirements, of missiles, pipe whip and discharging fluids - LR subsection 2.2.1, Pipe Rupture Locations and Associated Dynamic Effects; and LR subsection 2.5.1.3, Pipe Failures
- Feedwater flow control and isolation valve testing and valve closure requirements – LR subsection 2.2.4, Safety-Related Valves and Pumps
- Operation of the condensate and feedwater systems, including isolation features during postulated abnormal and accident scenarios is discussed in LR subsection 2.8.5, Accident and Transient Analyses and LR subsection 2.4.2 Plant Operability
- Feedwater isolation valve testing and valve closure, including containment isolation requirements – LR subsection 2.2.4, Safety-Related Valves and Pumps

2.5.5.4.2.3 Results

The following subsections evaluate the specific condensate and feedwater system capabilities while operating at SPU conditions.

General Design Criteria

The evaluation of the condensate and feedwater systems capabilities at SPU conditions demonstrates that CPNPP will continue to meet the current licensing basis with respect to the requirements of GDC-4, as described in LR subsection 2.2.1, Pipe Rupture Locations and Associated Dynamic Effects; and LR subsection 2.5.1.3, Pipe Failures.

GDC-5 is applicable to CPNPP as it is a dual unit installation. However, the feedwater and condensate systems are not shared between the units. CPNPP will continue to meet the current licensing basis with respect to the requirements of GDC-5.

The evaluation of the condensate and feedwater systems capabilities at SPU conditions demonstrates that CPNPP will continue to meet the current licensing basis with respect to the requirements of GDC-44, described in FSAR Section 3.1.4.15. Although the condensate and feedwater systems are not specifically addressed in GDC-44 or in FSAR Section 3.1.4.15, the feedwater system does provide an essential isolation function of feedwater flow to the steam generators and for containment isolation. The feedwater system provides this capability during accident conditions and is capable of achieving this function considering a single failure.

System Operating Conditions – Current Versus SPU Conditions

The condensate and feedwater system operating conditions; flow, temperature and pressure, were determined from hydraulic modeling of the piping systems and from the current operating (benchmark) and SPU heat balances. Tables 2.5.5.4-1 and 2.5.5.4-2 compares the current

condensate and feedwater system conditions to the SPU conditions for Units 1 and 2, respectively.

Design Pressures/Temperatures – Components and Piping

The design pressures and temperatures of condensate and feedwater components and piping bound the SPU operating conditions.

Feedwater Heaters

Feedwater heaters 1-1A/B through 1-6A/B and 2-1A/B through 2-6A/B were evaluated and determined acceptable for SPU operation based on their current design, materials, construction, and performance. Current plant operating and inspection data and the predicted SPU heat balance conditions have been reviewed to reach these conclusions. The industry criteria established by Heat Exchange Institute (HEI) standards have been generally used as the guidelines for acceptance, along with specific industry design guidelines.

Current plant operating data quantifying plant performance has been gathered and used to develop the current operating heat balance. This current operating heat balance was then adjusted to predict the plant performance at SPU conditions. These SPU heat balances show that the expected SPU power generation will be achieved thus confirming that the capability of the existing feedwater heaters is adequate for SPU operation.

The design and construction of the feedwater heaters is acceptable for continued operation at SPU conditions with specific monitoring programs in place to evaluate the potential for long-term degradation. The specifics of the plant monitoring program for erosion/corrosion effects are described in LR subsection 2.1.8, Flow Accelerated Corrosion. Other monitoring requirements are being implemented as noted below.

- Feedwater heater tube velocities are acceptable at SPU conditions.
- Feedwater heater tube side pressure drops are acceptable at SPU conditions.
- Feedwater heaters shell design pressures are acceptable for SPU conditions.
- The feedwater heaters shell and tube side relief valves were evaluated. All of the existing relief valve setpoints and capacities are acceptable for SPU operation.
- The feedwater heater shell side vents are acceptable for SPU operation.

Flow Velocities – Piping

Flow velocities through the condensate and feedwater system were calculated at current and SPU conditions. Flow velocities increased approximately 5 percent primarily due to the increased flow required by the SPU power level. Velocities remain below the maximum industry standard guidelines for these services although there are some pipes whose velocities exceed

the guidelines. Velocities at pumps and valves where increasers/reducers are located currently exceed the velocity guidelines. The SPU will result in slight increases to these existing velocities. These increasers/reducers are included in the Flow-Accelerated Corrosion (FAC) program and pipe wall thickness for these items has been acceptable. See LR subsection 2.1.8 for additional discussion.

Feedwater Flow Control Valves

The existing feedwater flow control valves (FCV-0510, 0520, 0530, 0540) provide the required flow at the required pressure drop at SPU conditions. The valve at SPU flow rates and increased feedwater pump speeds remains less than 80 percent open at SPU normal plant operation so as to provide sufficient control over a range of operating conditions.

The sizing and control capability of the feedwater flow control valves, together with the hydraulic operation of the feedwater pumps, provides sufficient flexibility to accommodate plant load rejection transients by providing 96 percent of rated flow with a 100 psi increase in steam generator pressure. This is consistent with Westinghouse Steam System Design criteria. For a large load rejection (up to 50%), the condensate pumps combined with the heater drain pumps can deliver the required 96 percent of rated flow and maintain the feedwater pump suction pressure above the SPU feedwater pump trip setpoint.

Condensate and Feedwater Pumps and Supporting Subsystems

The condensate pumps and feedwater pumps and their supporting subsystems will continue to operate successfully during SPU conditions based on the evaluation results, modifications and post-power-uprate inspections described below:

- The existing condensate pumps operate adequately at SPU with sufficient NPSH, flow, and head. The motors continue to provide sufficient power for pump operation at SPU conditions.

- The existing condensate pump recirculation system provides sufficient flow for condensate pump protection and supplies the minimum flow required by the gland steam condensers.

- The heater drain pump impellers are being replaced to ensure proper operation of the heater drain tank level control valve at the SPU. The motors continue to provide sufficient power for pump operation at SPU conditions.

- The heater drain pump minimum flow recirculation system provides sufficient flow for heater drain pump protection.

- The existing low pressure feedwater heater bypass and condensate polisher bypass lines will operate adequately at SPU conditions to provide sufficient flow and pressure at the feedwater pump suctions. The feed pump suction low pressure signal for the feedwater heater bypass valve and the condensate polishing bypass valve are being

modified to reduce the setpoint which maintains operating margin while ensuring adequate pump protection. Refer to LR subsection 2.4.1, Reactor Protection, Safety Features Actuation and Control Systems.

- The existing feedwater pump recirculation subsystem provides sufficient flow to meet the pump minimum flow requirements at SPU conditions.

- The existing seal water subsystem for the feedwater pumps and heater drain pumps continues to provide sufficient seal water flow and pressure from the condensate pumps. No changes are necessary to the seal water subsystem.

Feedwater Isolation Valves

Feedwater isolation is required for a variety of postulated transients and accident events. The current plant design provides for feedwater isolation using the main feedwater flow control valves, associated bypass valves, the feedwater isolation valves, and the feedwater isolation bypass valves. SPU pressure and temperature for these valves remain within the valves current design specification.

The containment isolation requirements are unaffected by SPU and the current plant design features remain acceptable.

The feedwater flow control valves and the associated bypass valves will experience a higher operating differential pressure at SPU conditions versus the current conditions. These valves remain within their current design conditions. Thus, these valves will continue to meet the required closure times.

Feedwater Check Valves

The feedwater check valves in each steam generator supply line prevent blowdown of the steam generators during a main feedwater line break in the common feedwater header. These valves remain within their current design conditions and, therefore, are acceptable for the SPU.

2.5.5.4.3 Conclusions

Luminant Power has reviewed the effects of the proposed SPU on the condensate and feedwater system and concluded that its assessment has adequately accounted for the effects of changes in plant conditions on the design of the condensate and feedwater system. It is concluded that the condensate and feedwater system, with the implementation of the above described modifications, and monitoring described above, will continue to maintain its ability to satisfy feedwater requirements for normal operation and shutdown, withstand water hammer, maintain isolation capability in order to preserve the system safety function, and not cause failure of safety-related SSCs. Luminant Power further concluded that the condensate and feedwater system will continue to meet the current licensing basis with respect to the requirements of GDC-4, GDC-5, and GDC-44. Therefore, the proposed SPU is acceptable with respect to the condensate and feedwater system.

Table 2.5.5.4-1 Condensate and Feedwater System Operating Conditions - Unit 1		
	Current Operating Condition	**SPU Operating Condition**
Condensate System		
Flow Rate, lbm/hr	9,879,419	10,356,060
Condenser Pressure, inches Hg @ Circ. Water Temperature, °F	2.74 @ 88.0°F CW	2.82 @ 88.0°F CW
Condensate Pump Discharge Pressure, psia	524.4	514.9
Condensate Supply Temperature, °F (FW Pump Suction)	351.6	353.9
Heater Drain System		
Heater Drain Pump Flow, lbm/hr	5,916,589	6,341,673
Heater Drain Pump Discharge Pressure, psia	413.5	404.2
Feedwater System		
Flow Rate, lbm/hr	15,508,480	16,322,730
Feedwater Pump Speed, rpm	4,975	5,100
Feedwater Pump Discharge Pressure, psia	1,248.3	1,266.0
Steam Generator Supply Temperature, °F	442.4	446.7

Table 2.5.5.4-2 Condensate and Feedwater System Operating Conditions - Unit 2		
	Current Operating Condition	**SPU Operating Condition**
Condensate System		
Flow Rate, lbm/hr	9,935,103	10,481,320
Condenser Pressure, inches Hg @ Circ. Water Temperature, °F	1.73 @ 71.0°F CW	1.78 @ 71.0°F CW
Condensate Pump Discharge Pressure, psia	523.3	508.0
Condensate Supply Temperature, °F (FW Pump Suction)	354.7	357.9
Heater Drain System		
Heater Drain Pump Flow, lbm/hr	5,705,109	5,914,271
Heater Drain Pump Discharge Pressure, psia	430.1	456.3
Feedwater System		
Flow Rate, lbm/hr	15,479,410	16,229,900
Feedwater Pump Speed, rpm	4,990	5,130
Feedwater Pump Discharge Pressure, psia	1,264.7	1,276.8
Steam Generator Supply Temperature, °F	441.0	444.5

2.5.6 Waste Management Systems

2.5.6.1 Gaseous Waste Management System

2.5.6.1.1 Regulatory Evaluation

Gaseous waste management systems involve the gaseous waste processing system, which deals with the management of radioactive gases collected in the waste gas decay tanks. In addition, it involves the management of the condenser air removal system and plant ventilation system exhausts.

The review focused on the effects that the proposed SPU may have on (1) the design criteria of the gaseous waste management systems, (2) methods of treatment, (3) expected releases, (4) principal parameters used in calculating the releases of radioactive materials in gaseous effluents, and (5) design features for precluding the possibility of an explosion if the potential for explosive mixtures exist. The Nuclear Regulatory Commission's (NRC's) acceptance criteria for the gaseous waste management systems are:

- 10 CFR 20.1302, insofar as it provides for demonstrating that annual average concentrations of radioactive materials released at the boundary of the unrestricted area do not exceed specified values.
- General Design Criterion (GDC) -3, insofar as it requires that:
  - Structures, systems, and components SSCs important to safety be designed and located to minimize the probability and effect of fires.
  - Noncombustible and heat-resistant materials be used.
  - Fire detection and fighting systems be provided and designed to minimize the adverse effects of fires on SSCs important to safety.
- GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents.
- GDC-61, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement.
- 10 CFR 50, Appendix I, Sections II.B, II.C and II.D, which set numerical guides for design objectives and limiting conditions for operation to meet the as-low-as-is-reasonably achievable (ALARA) criterion.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

The adequacy of the CPNPP design relative to conformance to:

- GDC-3 is described in FSAR Section 3.1.1.3, Criterion 3 – Fire Protection. As described in this FSAR section, SSCs are designed and located to minimize, consistent with other safety requirements, the probability and effects of fires and explosions. The fire fighting systems are designed to ensure that rupture or inadvertent operation does not significantly impair systems important to safety. The codes, standards, and regulations used in the design of the fire protection system and plant equipment are discussed in FSAR Section 9.5.1 and the Fire Protection Report.

- GDC-60 is described in FSAR Section 3.1.6.1, Criterion 60 – Control of Releases of Radioactive Materials to the Environment. As described in this FSAR section, waste handling systems are incorporated in the facility design for processing and/or retention of radioactive wastes for normal operation and anticipated operational occurrences. FSAR Section 11.3.1 states that the gaseous waste systems have sufficient capacity and redundancy to meet discharge concentration limits of 10 CFR 20 during periods of design basis fuel leakage, as discussed in FSAR Section 11.3.3. The gaseous waste processing system provides long-term holdup capacity, thus precluding the release of radioactive effluents during unfavorable environmental conditions. All gaseous effluent discharge paths are monitored for radioactivity, in compliance with GDC-64, as discussed in FSAR Section 11.5.

  FSAR Section 9.4 describes the primary plant ventilation system and non-engineered safety feature (ESF) exhaust units, which satisfy GDC-60. Chapter 11 describes the radioactive waste processing systems' design criteria, holdup capacities, and estimated releases of radioactive effluents to the environment. Compliance with 10 CFR 50, Appendix I, is described in Appendix 11A.

- GDC-61 is described in FSAR Section 3.1.6.2, Criterion 61 – Fuel Storage and Handling and Radioactivity Control. As described in this FSAR section, the radioactive waste processing systems, and other systems that contain radioactivity, are designed to assure adequate safety under normal and postulated accident conditions. The gaseous waste management systems are designed to ensure adequate safety under normal and postulated accident conditions by providing the following:

  - Components are designed and located such that appropriate periodic inspection and testing can be performed.

- All areas of the plant are designed with suitable shielding for radiation protection based on anticipated radiation dose rates and occupancy as discussed in FSAR Section 12.3.

- Individual components that contain significant radioactivity are located in confined areas, adequately ventilated through appropriate filtering systems.

- The primary plant ventilation system is designed to filter the exhaust from the radioactive waste and other systems which may contain radioactivity. The non-ESF exhaust units that satisfy GDC-61 are described in FSAR Section 9.4.

Additional details that define the licensing basis for the gaseous waste management systems are provided in FSAR Section 11.3.1. The gaseous waste systems include design features to appropriately monitor discharge streams and safety features to preclude releases in excess of 10 CFR 20 and to maintain radioactive discharges to levels ALARA according to the requirements of 10 CFR 50, Appendix I. The expected and design inventories of nuclides in the gaseous waste processing system components are provided in FSAR Section 11.3.2.1.4. Design provisions incorporated to control the release of radioactive materials in gaseous effluents and to maintain gas composition outside the range of flammable and explosive mixtures are described in FSAR Sections 11.3.2.1.1 and 11.3.2.1.2.

FSAR Chapter 15, Accident Analysis addresses a radioactive gas waste system leak or failure. FSAR Section 15.7.1 states that the whole body dose at the nearest point on the exclusion area boundary (EAB) is substantially below the 25 rem whole body value set forth in 10 CFR 100. The analysis concludes that such an event would not interrupt or restrict public use of areas beyond the EAB.

2.5.6.1.2 Technical Evaluation

2.5.6.1.2.1 Introduction

The gaseous waste management systems are described in FSAR Section 11.3, and have the capability to control, collect, process, store, and dispose of gaseous radioactive wastes generated from normal operation and anticipated operational occurrences. This is accomplished while achieving the lowest reasonable radioactive release to the environment available through current technology. The gaseous waste management systems include the gaseous waste processing system, ventilation systems, and the condenser evacuation (vacuum) system.

The gaseous waste processing system is shared between CPNPP Units 1 and 2. The main flow path in the gaseous waste processing system is a closed loop comprised of two waste gas compressors, two catalytic hydrogen recombiners, eight gas decay tanks for normal power service, and two gas decay tanks for service at shutdown and startup. The system also includes a gas decay tank drain pump, four gas traps, and a waste gas drain filter.

The gaseous waste processing system performs the following functions:

- To reduce fission product gas concentrations in the reactor coolant system (RCS) during normal operation
- To contain indefinitely the fission product gases removed from the RCS
- To collect gases generated by other systems
- To maintain a low level of hydrogen gas in the collected gases
- To discharge radioactive effluents after monitoring for radioactivity via a controlled path to ensure that the expected offsite doses are ALARA

The waste gases generated during plant operations including anticipated operational occurrences are collected and processed and are stored in waste gas decay tanks.

2.5.6.1.2.2 Description of Analyses and Evaluations

The gaseous waste management systems and components were evaluated to ensure they are capable of performing their intended functions at stretch power uprate (SPU) conditions. The evaluation considered existing design capabilities and whether SPU conditions would impact how gaseous radioactive wastes are collected, processed, stored and disposed of as a result of normal operation including anticipated operational occurrences.

Other evaluations are addressed in the following Licensing Report (LR) sections:

- LR subsection 2.5.3.2, Main Condenser Evacuation System
- LR subsection 2.7, Habitability, Filtration, and Ventilation
- LR subsection 2.10.1, Occupational and Public Radiation Doses

2.5.6.1.2.3 Results

The implementation of the SPU does not significantly increase the inventory of gas normally processed by the gaseous waste management systems since the plant system functions are not changing and the assumptions related to volume inputs remain the same. The SPU does not add or change any of the sources of potentially explosive mixtures.

Potentially radioactive gas is collected from selected systems and components and is directed to the gaseous waste processing system. The implementation of the SPU does not add any new sources of potentially contaminated gases, nor does it create any new flow paths or routes that would allow the contamination of uncontaminated gases.

Due to the increase in core power, the SPU results in an increase in the radioactivity in the reactor coolant, which impacts the concentrations of radioactive nuclides in the waste disposal systems. The radiological impact of the increased radioactivity with regards to exposure to plant

personnel and the public is detailed in LR subsection 2.10.1, Occupational and Public Radiation Doses.

Evaluation of the gaseous waste management systems at SPU conditions shows concurrence with 10 CFR 20.1302, insofar as the annual average concentrations of radioactive materials released at the boundary of the unrestricted area will not exceed specified values. This will be demonstrated by the continued compliance, post-SPU, to the annual dose objective of 10 CFR 50 Appendix I as discussed in LR subsection 2.10.1, Occupational and Public Radiation Doses. Discharge streams will remain appropriately monitored and adequate safety features remain incorporated to preclude excessive releases, in accordance with the offsite dose calculation manual.

Evaluation of the gaseous waste management systems at SPU conditions demonstrates that CPNPP will continue to meet the current licensing basis with respect to the requirements of GDC-3 in regards to fire detection and fighting systems. There is no impact to the fire detection or fighting systems due to SPU (LR subsection 2.5.1.4, Fire Protection). There are no new gaseous waste components added as a result of the SPU, and the gaseous waste system flow rates, gaseous inventory and process conditions remain within the original design parameters of the system. Thus, the existing systems retain their compliance to GDC-3.

Evaluation of the gaseous waste management systems at SPU conditions demonstrates that CPNPP will continue to meet the current licensing basis with respect to the requirements of GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents. This design capability remains unchanged by the SPU. The handling, control, and release of radioactive materials are in compliance with 10 CFR 50, Appendix I.

Evaluation of the gaseous waste management systems at SPU conditions demonstrates that CPNPP will continue to meet the current licensing basis with respect to the requirements of GDC-61, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement to ensure adequate safety under normal and postulated accident conditions. This design capability remains unchanged by the SPU.

The results of the postulated waste gas decay tank rupture/release remain within appropriate limits at SPU conditions. Refer to LR Section 2.9, Source Terms and Radiological Consequences Analyses, for details.

The evaluation of the gaseous waste management systems at SPU conditions demonstrates conformance with the requirements of 10 CFR 50, Appendix I, sections II.B, II.C and II.D, which set numerical guides for dose design objectives and limiting conditions for operation to meet the ALARA criterion. Refer to LR subsection 2.10.1, Occupational and Public Radiation Doses, for details.

SPU activities do not add any new components to the gaseous waste management systems, nor do they introduce any new functions for existing components. The changes associated with operating the gaseous waste management systems at SPU conditions do not add any new or previously unevaluated materials to the system.

2.5.6.1.3 Conclusions

The evaluation has confirmed that there is a negligible change in the amount of gaseous waste after the SPU and that the increase in fission products resulting from the increased equilibrium radioactivity of the RCS does not affect the ability of the gaseous waste management systems to process and control releases of radioactive materials and preclude the possibility of an explosion if the potential for explosive mixtures exists. The gaseous waste management systems will continue to meet their design functions and the requirements of 10 CFR 20.1302 and 10 CFR 50, Appendix I Sections II.B and II.C following implementation of the proposed SPU. CPNPP will continue to meet the current licensing basis with respect to the requirements of GDC-3, -60, and -61. The proposed SPU is acceptable with respect to the gaseous waste management systems.

2.5.6.2 Liquid Waste Management System

2.5.6.2.1 Regulatory Evaluation

The review of the liquid waste processing system focused on the effects that the proposed SPU may have on previous analyses and considerations related to the liquid waste processing system design, design objectives, design criteria, methods of treatment, expected releases, and principal parameters used in calculating the releases of radioactive materials in liquid effluents. The Nuclear Regulatory Commission (NRC) acceptance criteria for the liquid waste processing system are based on:

- 10 CFR 20.1302, insofar as it provides for demonstrating that annual average concentrations of radioactive materials released at the boundary of the unrestricted area do not exceed specified values.
- General Design Criterion (GDC)-60, insofar as it requires that the plant design include means to control the release of radioactive effluents.
- GDC-61, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement.
- 10 CFR 50, Appendix I, Sections II.A and II.D, which set numerical guides for dose design objectives and limiting conditions for operation to meet the "as-low-as-is-reasonably-achievable" criterion.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

The adequacy of the CPNPP design relative to conformance to:

- GDC-60 is described in FSAR Section 3.1.6.1, Criterion 60 - Control of Releases of Radioactive Materials to the Environment. As described in this FSAR section, waste handling systems are incorporated in the facility design for processing and/or retention of radioactive wastes for normal operation and anticipated operational occurrences. FSAR Section 11.2.1.1 states that the liquid waste processing system design ensures that quantities of radioactive releases to the environment meet the requirements specified in 10 CFR 20, 10 CFR 50, and the dose design objectives specified in Appendix I of 10 CFR 50, during both normal and anticipated operational occurrences.

  Control of liquid waste effluents is maintained by sampling and analyzing before discharge, and utilizing a controlled rate of release. A permanent record of radioactivity releases is provided by analyses of known volumes of effluent. Releases to the environment are monitored for radioactivity by radiation detectors.

  The liquid waste processing system is designed to handle occurrences of equipment faults of moderate frequency, such as a malfunction in the liquid waste processing system. The liquid waste processing system capacity is sufficient to handle expected transients in the processing of liquid waste.

- GDC-61 is described in the FSAR, Section 3.1.6.2, Criterion 61 – Fuel Storage and Handling and Radioactivity Control. As described in this FSAR section, radioactive waste processing systems, and other systems that contain radioactivity, are designed to assure adequate safety under normal and postulated accident conditions. Radioactive waste processing systems are designed to ensure adequate safety under normal and postulated accident conditions by providing the following:

  1. Components are designed and located such that appropriate periodic inspection and testing can be performed.

  2. All areas of the plant are designed with suitable shielding for radiation protection based on anticipated radiation dose rates and occupancy as discussed in FSAR Section 12.3.

  3. Individual components that contain significant radioactivity are located in confined areas, adequately ventilated through appropriate filtering systems. Radioactive waste management is discussed in detail in FSAR Chapter 11.

Additional information concerning the liquid waste processing system is provided in FSAR Section 11.2.

FSAR Chapter 15, Accident Analysis, addresses a radioactive liquid waste system leak or failure. FSAR Section 15.7.2 states that the thyroid and whole body doses at the exclusion area boundary are well within the limits set forth in 10 CFR 100.

FSAR Chapter 15, Accident Analysis, refers to postulated radioactive releases due to liquid tank failures, which are discussed in FSAR Sections 2.4.12 and 2.4.13.3. FSAR Section 2.4.12, Dispersion, Dilution, and Travel Time of Accidental Releases of Liquid Effluents in Surface Waters, provides a conservative analysis of a postulated accidental release of radioactive material in surface waters adjacent to the site due to the accidental rupture of a waste holdup tank. FSAR Section 2.4.13.3, Accident Effects, provides a conservative analysis of a postulated accidental release of radioactive liquid at the surface of the site due to an accidental rupture of the floor drain tank.

2.5.6.2.2 Technical Evaluation

2.5.6.2.2.1 Introduction

The liquid waste processing system is described in FSAR Section 11.2, and has the capability to control, collect, process, handle, store, and dispose of liquid radioactive wastes generated from normal operation and anticipated operational occurrences. This is accomplished while achieving the lowest reasonable radioactive release to the environment available through current technology.

The liquid waste processing system services both CPNPP Units 1 and 2 with shared components. The liquid waste processing system is designed to segregate different effluents from equipment leaks and drains according to their chemical and radiochemical properties. The system is divided into the following subsystems:

- Reactor coolant drain subsystem
- Drain Channels A, B, and C
- Spent resin handling subsystem
- Filter demineralizer subsystem

The liquid waste processing system is designed to perform the following functions within the subsystems listed above:

- Collect, process, and recycle reactor-grade effluents, including equipment leaks and drains, valve leakoffs, pump seal leakoffs, loop drain leakoffs, and other tritiated waste sources

- Collect and process floor drains and laundry waste, including equipment drains containing non-reactor grade water, controlled area sink drains, laundry, personnel decontamination showers and sinks, surface decontamination wastes and other non-reactor grade drain sources, and recycle or discharge the processed floor drain and laundry waste

- Remove and concentrate radioactive constituents, and process concentrated radioactive constituents for solidification and shipment offsite

Liquid waste is released primarily on a batch basis to permit optimum control and disposal. Fluids are sampled and analyzed prior to being discharged. Based on this analysis, the wastes are retained for further processing or released under controlled conditions via the circulating water discharge canal. A radiation monitor will automatically terminate the liquid waste discharge when the waste activity reaches the monitor setpoint.

Waste disposal requirements are also discussed in FSAR Section 9.3.3, Equipment and Floor Drainage System.

2.5.6.2.2.2 Description of Analyses and Evaluations

The liquid waste processing system and components were evaluated to ensure they are capable of performing their intended functions at SPU conditions. The evaluation considered existing design capabilities and whether SPU conditions would impact how liquid radioactive wastes are controlled, collected, processed, handled, stored and disposed of as a result of normal operation including anticipated operational occurrences.

Other evaluations are addressed in the following Licensing Report (LR) sections:

- LR subsection 2.10.1, Occupational and Public Radiation Doses

2.5.6.2.2.3 Results

The implementation of SPU does not significantly increase the inventory of liquid normally processed by the liquid waste processing system since the system functions are not changing and the assumptions related to volume inputs remain the same. The steam generator blowdown flow rates are not impacted by the SPU (LR subsection 2.1.10).

Potentially radioactive drainage is collected in tanks and drain sumps from selected systems and components and is directed to the appropriate radwaste processing subsystem. Liquids leaking from process systems, liquids used during cleaning activities, and liquid spills from maintenance activities enter the equipment and floor drain system during all plant operating modes. The implementation of SPU does not add any new sources of potentially contaminated leakage, nor does it create any new flow paths or routes that would allow the contamination of drainage systems designed for uncontaminated fluids.

Due to the increase in core power, the SPU results in an increase in radioactivity in the reactor coolant which impacts the concentrations of radioactive nuclides in the waste disposal systems. The radiological impact of the increased radioactivity with regards to personnel and public exposure is detailed in LR subsection 2.10.1, Occupational and Public Radiation Doses.

The evaluation of the liquid waste processing system at SPU conditions shows conformance with 10 CFR 20.1302, insofar as the annual average concentrations of radioactive materials released at the boundary of the unrestricted area will not exceed specified values. This will be demonstrated by the continued compliance, post-SPU, to the annual dose objective of 10 CFR 50 Appendix I as discussed in LR subsection 2.10.1, Occupational and Public Radiation

Doses. Discharge streams will remain appropriately monitored and adequate safety features remain incorporated to preclude excessive releases.

The evaluation of the liquid waste processing system at SPU conditions demonstrates that CPNPP will continue to meet the current licensing basis with respect to the requirements of GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents. This design capability remains unchanged by the SPU. Thus, the handling, control, and release of radioactive materials will remain in compliance with 10 CFR 50, Appendix I.

The evaluation of the liquid waste processing system at SPU conditions demonstrates that CPNPP will continue to meet the current licensing basis with respect to the requirements of GDC-61, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement to ensure adequate safety under normal and postulated accident conditions. This design capability remains unchanged by the SPU.

The evaluation of the liquid waste processing system at SPU conditions demonstrates conformance with the requirements of 10 CFR 50, Appendix I, Sections II.A and II.D, which set numerical guides for dose design objectives and limiting conditions for operation to meet the "as-low-as-is-reasonably-achievable" criterion. Refer to LR Section 2.10.1, Occupational and Public Radiation Doses, for details.

The results of a postulated radioactive liquid waste system leak or failure remain within appropriate limits at SPU conditions. Refer to LR Section 2.9.12, Radiological Consequences of Liquid Waste Tank Rupture, for details.

The liquid waste processing system flow rates, water inventory and process conditions are not changed by the SPU, and there are no system or component modifications necessary. SPU activities do not add any new components to the liquid waste processing system, nor do they introduce any new functions for existing components that would change the license renewal system evaluation boundaries. The changes associated with operating the liquid waste processing system at SPU conditions do not add any new or previously unevaluated materials to the system.

2.5.6.2.3 Conclusions

The evaluation has confirmed that there is a negligible change in the amount of liquid waste after the SPU and that the increase in fission products does not affect the ability of the liquid waste processing system to control releases of radioactive materials. The liquid waste processing system will continue to meet its design functions and the requirements of 10 CFR 20.1302 and 10 CFR 50, Appendix I, Sections II.A and IID. CPNPP Units 1 and 2 will continue to meet the current licensing basis with respect to the requirements of GDC-60 and GDC-61. Therefore, the proposed SPU is acceptable with respect to the liquid waste processing system.

2.5.6.3 Solid Waste Management System

2.5.6.3.1 Regulatory Evaluation

The review of the solid waste management system focused on the effects that the proposed SPU may have on previous analyses and considerations related to the design objectives in terms of expected volumes of waste to be processed and handled, the wet and dry types of waste to be processed, the activity and expected radionuclide distribution contained in the waste, equipment design capacities, and the principal parameters employed in the design of the solid waste management systems. The Nuclear Regulatory Commission's (NRC's) acceptance criteria for the solid waste management system are based on:

- 10 CFR 20.1302, insofar as it provides for demonstrating that annual average concentrations of radioactive materials released at the boundary of the unrestricted area do not exceed specified values.
- General Design Criterion (GDC)-60, insofar as it requires that the plant design include a means to control the release of radioactive effluents.
- GDC-63, insofar as it requires that systems be provided in waste handling areas to detect conditions that may result in excessive radiation levels.
- GDC-64, insofar as it requires that a means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and postulated accidents.
- 10 CFR 71, which states requirements for radioactive material packaging.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

The adequacy of the CPNPP design relative to conformance to:

- GDC-60 is described in the FSAR Section 3.1.6.1, Criterion 60 – Control of Releases of Radioactive Materials to the Environment. As described in this FSAR section, waste handling systems are incorporated in the facility design for processing and/or retention of radioactive wastes for normal operation and anticipated operational occurrences.

  Solid wastes are prepared for offsite disposal by either compaction or solidification. Bulk disposal of wastes is accomplished via truck-mounted or mobile waste processing systems. Compressible low-radiation-level solid wastes are processed onsite using the

waste baling system, or shipped to an offsite vendor for immediate processing and volume reduction prior to disposal. The baler's shroud is ducted to the plant ventilation system to remove dust or particles that may be emitted from the drum during compression of the wastes. Incompressible solids are packaged in suitable containers.

FSAR Section 9.4 describes the primary plant ventilation system and non-engineered safety features exhaust units which satisfy GDC-60. Chapter 11 describes the radioactive waste processing systems' design criteria, holdup capacities, and estimated releases of radioactive effluents to the environment. Compliance with 10 CFR 50, Appendix I, is described in Appendix 11A.

- GDC-63 is described in FSAR Section 3.1.6.4, Criterion 63 – Monitoring Fuel and Waste Storage. As described in this FSAR section, monitoring systems and alarms are provided as required to warn personnel of impending excessive levels of radiation or airborne activity, or to alarm on excessive temperature or low water level in the spent fuel pool. Appropriate safety actions will be initiated by operator action.

- GDC-64 is described in FSAR Section 3.1.6.5, Criterion 64 – Monitoring Radioactivity Releases. As described in this FSAR section, radioactivity levels in the effluent discharge paths and in the plant environs are continually monitored during normal and accident conditions by the process radiation monitoring system as described in FSAR Section 11.5. Provisions for monitoring plant areas are included in the area radiation monitoring system described in FSAR Section 12.3.4. Periodic surveys are conducted using portable equipment as discussed in FSAR Section 12.5.

Additional details that define the licensing basis for the solid waste management system are described in FSAR Section 11.4. Wastes processed via truck-mounted or mobile waste processing systems are supplied from the chemical drain tank, waste conditioning tank, the nuclear steam supply system (NSSS) spent resin transfer system, and the steam generator blowdown spent resin transfer system. Compressible low-radiation-level solid wastes processed via the WBS are products of plant operation and maintenance. The WBS waste baler meets the requirements of 10 CFR Parts 20, 50, 61 and 71 and United States Department of Transportation (DOT) Hazardous Materials Regulation 49 CFR Parts 170 through 178.

The solid waste management system product is a container of radioactive material packaged in accordance with all applicable NRC, DOT, and disposal site requirements. Shielded containers are used when necessary for higher radiation level waste containers to ensure radiation levels at and beyond the staging and handling area fence are within 10 CFR 20 limits for unrestricted areas. All radioactive waste shipments are in compliance with the applicable regulatory standards and requirements of the NRC, DOT, Texas Regulations for Radiation Control, and the burial site.

2.5.6.3.2 Technical Evaluation

2.5.6.3.2.1 Introduction

The solid waste management system is described in FSAR Section 11.4, and has the capability to control, collect, condition, handle, process, package, and temporarily store, prior to offsite shipment, solid radioactive waste generated from normal operation, including anticipated operational occurrences. The types of solid waste produced at CPNPP include dry active waste, evaporator concentrates, resins, sludges from tanks and sumps, and spent filter cartridges.

The solid waste management system design functions are to:

- Provide a means of encapsulating or compacting radioactive solid wastes generated by reactor plant operations
- Provide adequate equipment and storage area shielding for the protection of operating personnel pending shipment of waste to disposal facilities
- Measure and record the radiation levels of the solid waste processed for shipment from the site to disposal facilities
- Provide a 3-month to 6-month storage capacity for the processed wastes depending upon plant operation

2.5.6.3.2.2 Description of Analyses and Evaluations

The solid waste processing system and components were evaluated to ensure they are capable of performing their intended functions at SPU conditions. The evaluation considered existing design capabilities and whether SPU conditions would impact how solid radioactive wastes are controlled, collected, conditioned, handled, processed, packaged, and temporarily stored, prior to offsite shipment, as a result of normal operation including anticipated operational occurrences.

2.5.6.3.2.3 Results

The proposed SPU has no significant effect on the generation of solid waste volume from the primary and secondary systems since the system functions are not changing and the assumptions related to volume inputs remain the same.

Due to the increase in core power, the SPU results in an increase in the radioactivity in the reactor coolant which impacts the concentrations of radioactive nuclides in the waste disposal systems. The impact of the increased radioactivity with regards to exposure to plant personnel and the public is detailed in Licensing Report (LR) subsection 2.10.1, Occupational and Public Radiation Doses.

The evaluation of the solid waste management system at SPU conditions demonstrates concurrence with 10 CFR 20.1302 since the annual average concentrations of radioactive materials released at the boundary of the unrestricted area will not exceed specified values. This is demonstrated by the continued compliance, post-SPU, to the annual dose objective of 10 CFR 50 Appendix I as discussed in LR subsection 2.10.1, Occupational and Public Radiation Doses. Discharge streams will remain appropriately monitored and adequate safety features remain incorporated to preclude excessive releases.

The evaluation of the solid waste management system at SPU conditions demonstrates that CPNPP will continue to meet the current licensing basis with respect to the requirements of GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents. This design capability remains unchanged by the SPU. The handling, control, and release of radioactive materials will remain in compliance with 10 CFR 50, Appendix I.

The evaluation of the solid waste management system at SPU conditions demonstrates that CPNPP will continue to meet the current licensing basis with respect to the requirements of GDC-63, insofar as it requires that systems be provided in waste handling areas to detect conditions that may result in excessive radiation levels and to initiate appropriate safety actions. This design capability remains unchanged by the SPU. Radiation monitors and alarms are provided as required to warn personnel of impending excessive levels of radiation or airborne activity. Refer to LR subsection 2.10.1, Occupational and Public Radiation Doses.

The evaluation of the solid waste management system at SPU conditions demonstrates that CPNPP will continue to meet the current licensing basis with respect to the requirements of GDC-64, insofar as it requires that a means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and postulated accidents. This design capability remains unchanged by the SPU. Radioactivity levels contained in the effluent discharge paths in the environs are continually monitored during normal and accident conditions by the process and effluent radiological monitoring and sampling systems. Refer to LR subsection 2.10.1, Occupational and Public Radiation Doses.

The evaluation of the solid waste management system at SPU conditions demonstrates conformance with the requirements of 10 CFR 71, insofar as the radioactive material packaging accounts for the maximum dose rate allowed on the surface of the container by shielding of the package in which the container is shipped. The requirements for packaging, shielding, handling and shipping of radioactive materials are not changed by SPU; thus, compliance with 10 CFR 71 is not affected.

SPU activities do not add any new components to the solid waste management system, nor do they introduce any new functions for existing components. The changes associated with operating the solid waste management system at SPU conditions do not add any new or previously unevaluated materials to the system.

The increase in other solid wastes generated at SPU conditions is considered insignificant relative to pre-SPU levels as SPU will not significantly impact installed equipment operation or maintenance.

Implementation of SPU is anticipated to increase the potential for occurrence of the crud-induced power shift (CIPS) phenomena. Luminant Power currently manages/precludes the occurrence of CIPS by managing reactor water chemistry in conjunction with the design of the reactor core. Following implementation of SPU, occurrence of the CIPS phenomena will continue to be managed/precluded by managing reactor water chemistry during plant operation.

2.5.6.3.3 Conclusions

The SPU has an insignificant impact on the solid waste management system. No modifications to the solid waste management system are required for SPU. The effect of the increase in fission product and amount of solid waste on the ability of the solid waste management system to process the waste has been evaluated and the solid waste management system meets its design functions following implementation of the proposed SPU. The solid waste management system continues to meet the requirements of 10 CFR 20.1302 and 10 CFR 71. CPNPP Units 1 and 2 will continue to meet the current licensing basis with respect to the requirements of GDC-60, -63, and -64. Therefore, the proposed SPU is acceptable with respect to the solid waste management system.

2.5.7 Additional Considerations

2.5.7.1 Emergency Diesel Engine Fuel-Oil Storage and Transfer System

2.5.7.1.1 Regulatory Evaluation

Nuclear power plants are required to have redundant onsite emergency power supplies of sufficient capacity to perform their safety functions (such as power-diesel engine-driven generator sets), assuming a single failure. This review focused on increases in emergency diesel generator (EDG) electrical demand and the resulting increase in the amount of fuel oil necessary for the system to perform its safety function.

Current Licensing Basis

Final Safety Analysis Report (FSAR) Section 9.5.4, states the EDG fuel-oil storage and transfer system is designed to supply a reliable source of fuel oil for the four EDGs (two per unit) continuous operation at rated load for a period of not less than seven days. The EDG fuel-oil storage and transfer system design is in accordance with the following criteria from 10 CFR 50, Appendix A:

- General Criterion (GDC) -2, Design Bases for Protection Against Natural Phenomena
- GDC-4, Environmental and Missile Design Bases
- GDC- 5, Sharing of Systems, Structures, and Components

The system also complies with the following guidance and standards as described in the FSAR:

- NRC Regulatory Guide 1.29, Seismic Design Classification
- NRC Regulatory Guide 1.137, Fuel-Oil Systems for Standby Diesel Generators (1/1978)
- The criteria for protection against pipe break outside the containment conforms to the guidelines contained in Branch Technical Position (BTP) APCSB 3-1 and BTP 3-1 as described in FSAR, Section 3.6

2.5.7.1.2 Technical Evaluation

2.5.7.1.2.1 Introduction

FSAR Section 8.3.1.2.1, Compliance, indicates there are two independent EDGs per unit to insure the compliance with 10 CFR 50, Appendix A, GDC-17. They have equal capacity, characteristics, and interchangeable components. An independent fuel-oil storage and transfer system and distribution system is provided for each diesel generator set.

The fuel-oil storage and transfer system described in FSAR Section 9.5.4 provides and maintains a minimum inventory of diesel fuel oil to ensure that both EDGs for each unit can operate at their design ratings for seven days. The fuel-oil storage and transfer system for each

engine is separate and independent of the others. Each flow path consists of a fuel oil storage tank, two 125-percent capacity fuel oil transfer pumps, a strainer, a day tank, and piping to each respective diesel engine. Each day tank has one supply and one return connection to the fuel oil injection system, mounted integrally, and provided with its respective diesel engine.

The EDG system and fuel-oil storage and transfer system are designed to meet single failure criterion, to withstand the effects of the worst anticipated environmental phenomenon, and to meet seismic Category I requirements. The EDG fuel-oil storage and transfer system is designed to Institute of Electrical and Electronics Engineers (IEEE) 387 and seismic Category I requirements.

The EDG system and fuel-oil storage and transfer system are designed to conform to Regulatory Guide 1.137 with exceptions as described in FSAR Appendix 1A(B).

The fuel-oil day tank provides an immediate and sufficient source of fuel oil for the EDG to run for a minimum of a one hour at rated load plus a minimum margin of 10 percent per American National Standards Institute (ANSI) N195. The fuel-oil day tank is located in the diesel generator room, is enclosed by three hour fire-walls and is elevated above the diesel generator to maintain a positive pressure at the suction of the engine fuel pumps.

The fuel-oil transfer pump is to transfers fuel oil from the storage tank to the day tank and has sufficient capacity to fill the day tank with the EDG running at rated load and speed.

2.5.7.1.2.2 Description of Analyses and Evaluations

To ensure the functionality of the EDG system subsequent to implementation of the stretch power uprate (SPU), the quantity of fuel oil, fuel consumption rate, and lube oil requirements were evaluated.

The EDG fuel-oil and transfer system and its components were evaluated to ensure they are capable of performing their intended function at SPU. The evaluation is based on the system's required design functions and a comparison between the existing equipment ratings and the anticipated operating requirements at SPU.

There are no changes in EDG loading or run times as a result of SPU.

2.5.7.1.2.3 Results

The NSSS loads, turbine-generator (TG) and balance-of-plant (BOP) loads were reviewed for increases in load or duration of required operation. There are no changes in the loads connected to the electrical safety busses. EDG loading and run times will not change at SPU conditions.

The EDG fuel-oil and transfer system requirements for operation at SPU conditions remain bounded by current analysis.

The EDG and auxiliaries are reviewed in Licensing Report (LR) subsection 2.3.3.

The quantity of fuel oil required is bounded by Technical Specification limits, and fuel consumption rates are not affected for EDG operation at SPU conditions.

2.5.7.1.3 Conclusion

The EDG fuel-oil and transfer system will function as designed and meet the current licensing basis following implementation of the proposed SPU. The fuel-oil and transfer system will provide an adequate amount of fuel oil to allow the EDGs to meet the onsite power requirements at SPU conditions.

2.5.7.2 Light Load Handling System (Related to Refueling)

2.5.7.2.1 Regulatory Evaluation

The light load handling system (LLHS) includes components and equipment used in handling new fuel at the receiving station and the loading of spent fuel into shipping casks. New fuel assemblies received for refueling are removed one at a time from the shipping container and moved to the new fuel assembly inspection area. Fuel is moved between the reactor vessel and the containment fuel transfer area by the refueling machine. The fuel transfer system (FTS) is used to move fuel assemblies between the Containment Building and the Fuel Building.

Luminant Power review covered the avoidance of criticality accidents, radioactivity releases resulting from damage to irradiated fuel, and unacceptable personnel radiation exposures. The review focused on the effects of the fuel on system performance and related analysis. The Nuclear Regulatory Commission's (NRC's) acceptance criteria for the LLHS are based on:

- General Design Criterion (GDC) -61, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement and with suitable shielding for radiation protection.

- GDC-62, insofar as it requires that criticality be prevented.

Current Licensing Basis

The Comanche Peak Nuclear Power Plant's (CPNPP's) control of new and spent fuel movement is as follows:

- GDC-61 is described in Final Safety Analysis Report (FSAR) Section 3.1.6.2, Fuel Storage and Handling and Radioactivity Control. As described in this FSAR section, the fuel handling system is designed to ensure adequate safety under normal operation and postulated accident conditions. The fuel handling system is FSAR Section 9.1.4, Fuel Handling Systems.

- GDC-62 is described in FSAR Section 3.1.6.3, Prevention of Criticality in Fuel Storage and Handling. As described in this FSAR section, criticality in new and spent fuel storage areas is prevented both by physical separation of fuel assemblies and by the presence of borated water in the spent fuel storage pool. The fuel handling system is discussed in FSAR Section 9.1.4, Fuel Handling Systems.

2.5.7.2.2 Technical Evaluation

The fuel handling equipment is designed to handle the spent fuel assemblies underwater from the time they leave the reactor vessel until they are placed in a container for shipment from the site. Underwater transfer of spent fuel assemblies provides an effective, economic, and transparent radiation shield, as well as a reliable cooling medium for removal of decay heat.

Component descriptions are contained in FSAR Section 9.1.4.

Results

The fuel type is mechanically the same as has been used at CPNPP Units 1 and 2. As such, the weight of the assemblies has not changed. Therefore, the bridge crane, overhead crane, and fuel transfer system remain qualified to perform their functions at uprate conditions.

2.5.7.2.3 Conclusions

This review finds that there are no effects of the fuel on the ability of the LLHS to avoid criticality accidents since the fuel type does not change for SPU. Based on this review, Luminant Power concludes that the LLHS will continue to meet the CPNPP current licensing basis with respect to the requirements of GDC-61 and GDC-62 for radioactivity releases and prevention of criticality accidents. Therefore, Luminant Power finds the proposed SPU acceptable with respect to the LLHS.

2.5.8 Additional Review Areas (Plant Systems)

2.5.8.1 Circulating Water System

2.5.8.1.1 Regulatory Evaluation

The circulating water (CW) System provides a continuous supply of cooling water to remove the heat rejected by the turbine cycle and the auxiliary systems. The review of the system focused on the capability of the current design CW flow to remove the increased amount of heat loads to the condensers and other turbine cycle heat exchangers due to the stretch power uprate (SPU). The impact of this increased heat on the CW components is evaluated to ensure that the system accomplishes its design functions after implementation of the SPU.

Specific CW flow and discharge temperature limits to the Squaw Creek Reservoir (SCR) are contained in the Texas Pollutant Discharge Elimination System (TPDES) permit. The current plant design to prevent safety-related facilities from flood damage due to a CW pipe break is evaluated in LR subsection 2.5.1.1.3.

Current Licensing Bases

The Comanche Peak Nuclear Power Plant (CPNPP) Final Safety Analysis Report (FSAR) sections that address the CW system include:

- FSAR Section 10.4.5 – This section addresses CW system descriptions, design bases, CW heat sink, safety evaluation, inspection and testing requirements, and instrumentation requirements.

  The CW system is not required for emergency cooldown or for operation of the engineered safeguard system; instead, the station service water system and the related safe shutdown impoundment (SSI) fulfill these functions.

  The CW system is not required for cooling during shutdown because reactor coolant system (RCS) cooldown can be accomplished by dumping steam to the atmosphere through the power-operated main steam relief valves.

- FSAR Section 11.5 – Process and effluent radiological monitoring and sampling systems. The last part of this section describes that liquid can be discharged to the CW discharge tunnel after it is processed by the liquid waste processing system, and will be monitored by a gamma-sensitive detector before release to the CW discharge tunnel. The Turbine Building sump effluent is monitored by gamma-sensitive monitors prior to release to the low volume waste pond. Upon detection of radioactivity, this effluent is diverted to the waste water holdup tanks where it is sampled and, if appropriate, released from the holdup tanks on a batch basis to the CW discharge tunnel.

- FSAR Section 2.4.2 – This section describes the historical maximum flood in the rivers near SCR, the causes of flood, the flood design considerations, and the magnitude of the probable maximum precipitation (PMP).

- Section 2.4.4 – States that a seismically induced dam failure of the Squaw Creek dam presents no danger of flooding the CPNPP because the station is above the PMP water level. Floods are also discussed in FSAR Appendix 1A (B), Regulatory Guideline 1.102.

2.5.8.1.2 Technical Evaluation

2.5.8.1.2.1 Introduction

The CW system is discussed in the FSAR Sections 10.4.5. The main function of the CW System is to provide a reliable supply of cooling water to condense the steam exhausted from the low-pressure turbines and the feedwater pump turbines. The CW system also provides cooling water to selected chillers and miscellaneous heat exchangers in the turbine plant.

The CW piping delivers cooling water from the SCR and conveys the heated discharge water back to the SCR. Each of the CW systems in Unit 1 and Unit 2 consist of CW pumps, trash racks, traveling screens, expansion joints, condensers, and heat exchangers. The principal structures of the CW systems are intake structure, intake and discharge tunnels, and the discharge structure.

Approximately 94 percent of the CW pump flow distributes to the tube side of the main condensers to remove the heat rejected by the turbine cycle. Approximately 4 percent of the flow distributes to the tube side of the auxiliary condenser to cool the turbine exhaust of the feedwater pump. This evaluation focuses on the increased amount of rejected heat to be absorbed by CW system and the increased discharge temperature. The shell side of the condenser is evaluated in LR subsection 2.5.5.2.

TPDES regulates both the flow rate and temperature of CW discharged into the SCR.

2.5.8.1.2.2 Description of Analyses and Evaluations

The CW system and its components were evaluated to ensure they are capable of performing their intended function at the SPU operating conditions. The evaluation reviewed the CW system to determine whether the existing CW flow rate is capable of removing the higher steam cycle rejected heat at SPU conditions. The evaluation also reviewed the auxiliary condensers of the feedwater pump turbines and the heat exchangers and coolers in the turbine plant to ensure that the current CW is adequate to remove the uprated heat load.

A hydrothermal mathematical model, developed specifically for the SCR, simulates the thermal conditions in the reservoir. The model is used to predict the water temperature and the evaporation rate in the SCR.

The increased heat to the CW system from the turbine cycle heat loads at SPU conditions raises the system operating CW supply temperature to the main condenser and increases the CW outlet temperature of the main condenser and downstream of the other affected heat exchangers. Heat loads during normal plant operation with different cooling water temperatures are utilized in the heat balance studies for the evaluation. The predicted increase in CW supply temperature due to the uprated power level is less than 1°F. The heat balances for the uprated condition were performed using a 102°F, which bounds the expected SCR temperature at uprate conditions. The resulting increase in the discharge CW temperature is approximately 1.5°F.

The heat balances demonstrate that there is sufficient heat removal capability with the existing CW flow at the supply temperature of 102°F. The small increase in the CW temperature does not have an adverse impact on the existing component design parameters and functions. The higher circulating water discharge temperatures were also reviewed against the TPDES permit.

Circulating water at CPNPP is drawn from the SCR. SPU does not affect the existing chemical treatment program employed at CPNPP. All of the existing limitations on discharge to the SCR will be maintained for SPU in accordance with the TPDES permit.

As permitted based on routine sampling and analyses, liquid waste is periodically released, under controlled conditions, via the CW discharge tunnel. As detailed in LR subsection 2.5.6.2, evaluation of liquid waste processing system at SPU conditions demonstrates that the change in the amount of liquid waste is negligible, that CPNPP will continue to meet its licensing basis, and that the design capability remains unchanged in conjunction with system releases. Therefore, there is no impact on the CW system relative to the effects of liquid waste discharge to circulating water flow at the discharge tunnel (that is, dilution characteristics are unchanged).

Other evaluations related to the CW System, piping and components are included in the following LR sections:

- Subsection 2.5.6.2, Liquid Waste Management System
- Subsection 2.5.1.1, Flooding
- Subsection 2.2.2, Pressure-Retaining Components and Component Supports
- Subsection 2.5.2.2, Main Condenser

Also see the Environment Impact Assessment Report.

2.5.8.1.2.3 Results

Normal operation – The SPU heat removal capability of the main condenser is adequate based on current condenser CW flows and a maximum CW supply temperature of 102°F. The resultant maximum condenser pressure is approximately 4.09 in Hg (as compared with current conditions of approximately 3.97 in Hg at the maximum CW supply temperature of 102°F). No physical changes are required in the CW system. The current CW flow rate is acceptable for the SPU conditions. For specifics on the main condenser refer to LR subsection 2.5.5.2.

The small increase in the CW temperature does not have an adverse impact on the existing component design parameters and functions.

SPU operating pressures within the system do not change since the current CW flow rates are acceptable and the circulating pumps will continue to operate at the same flow and discharge head at SPU conditions. The existing piping and the component design pressures are unaffected by the SPU conditions.

SCR temperature – The CW supply temperature is predicted to increase less than 1°F due to uprate. Current practice to lower the temperature of the SCR is to pump water into SCR from Lake Granbury. The pumping flow replaces the reservoir natural evaporation with lower temperature water. The pumping also initiates a spillway overflow that discharges high-temperature water in the top layer of the SCR. The pumping helps in reducing the CW discharge temperature during the summer-time operation.

TPDES discharge permit – The TPDES permit allowance is for a CW flow rate of 3,168 mgd (2,200,000 gpm). The daily average discharge temperature is 113°F and the daily maximum is 116°F. The total CW flow rate for Unit 1 and Unit 2 is unchanged at the SPU operation and therefore, the permit allowable flow rate limit is unaffected by the SPU.

The CW discharge temperature is monitored at the outlet of the CW discharge structure. The results of the heat balance studies show that, at the SPU operating condition, the discharge CW temperature will increase approximately 1.5°F. The expected increase, however, will be within the limits specified in the CPNPP TPDES wastewater discharge permit under normal conditions. Luminant Power will comply with the stations TPDES permit limitations if the CW outlet discharge temperature encroaches on the permitted limits, even during atypical severe environmental conditions.

2.5.8.1.3 Conclusions

The effects of the proposed SPU on the CW system have been evaluated. The evaluation concludes that the current CW system will be adequate and accounts for the effects of the proposed SPU on the system's capability to remove heat rejected from the turbine cycle and auxiliary heat exchangers. The current design of the CW system provides a reliable supply of water at SPU conditions to condense the steam exhausted from the low-pressure turbines and steam generator feedwater pump turbines, and to remove heat from miscellaneous heat exchangers. The current design of the CW system piping and its components accommodates the higher condenser duty and higher temperatures at SPU conditions. The TPDES thermal discharge limits are not required to change.

2.6 CONTAINMENT REVIEW CONSIDERATIONS

2.6.1 Primary Containment Functional Design

2.6.1.1 Regulatory Evaluation

The containment encloses the reactor system and is the final barrier against the release of significant amounts of radioactive fission products in the event of an accident.

The Luminant Power review covered the pressure and temperature conditions in the containment due to a spectrum of postulated loss-of-coolant accidents (LOCAs) and secondary system line breaks.

The Nuclear Regulatory Commission (NRC) acceptance criteria for primary containment functional design are based on:

- General Design Criterion (GDC)-16, insofar as it requires that reactor containment be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment.
- GDC-50, insofar as it requires that the containment and its internal components be able to accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA.
- GDC-38, insofar as it requires that the containment heat removal system(s) function to rapidly reduce the containment pressure and temperature following any LOCA and maintain them at acceptably low levels.
- GDC-13, insofar as it requires that instrumentation be provided to monitor variables and systems over their anticipated ranges for normal operation and accident conditions.
- GDC-64, insofar as it requires that means be provided for monitoring the plant environs for radioactivity that may be released from normal operations and postulated accidents.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the adequacy of the design relative to:

- GDC-16, Containment Design, is described in FSAR Section 3.1.2.7.

  A steel-lined reinforced concrete containment structure encloses the entire reactor coolant system (RCS) and is designed to withstand the pressures and temperatures resulting from a spectrum of postulated LOCAs and secondary system breaks.

  The emergency core cooling system, (ECCS) cools the reactor core and limits the release of radioactive materials to the environment.

  To ensure its integrity, the containment spray system (CT) and hydrogen removal system are incorporated in the containment design. The CT is designed to function after a LOCA to reduce the pressure inside the containment to near atmospheric pressure and to remove fission product activity from the containment atmosphere.

  .The hydrogen purge system is designed to prevent hydrogen gas from reaching a combustible concentration in the Containment Building as specified in NRC Regulatory Guide 1.7, Control of Combustible Gas Concentration in the Containment Following a Loss-of-Coolant Accident.

  The Containment Building and engineered safety feature (ESF) systems are designed to safely sustain internal and external environmental conditions that may reasonably be expected to occur during the life of the plant, including both short-and long-term effects following a LOCA (See FSAR Sections 6.2, 6.5, 15.6, and 3.8.1)

- GDC-50, Containment Design Basis, is described in FSAR Section 3.1.5.1.

  The Containment Building is designed to withstand pressures and temperatures resulting from a spectrum of LOCAs and secondary system ruptures.

  Vital Containment Building subcompartments, such as the steam generator compartment, the reactor cavity, and the pressurizer compartment, are deigned to withstand, with a safe margin, peak differential pressures resulting from postulated hot leg, cold leg, and pressurizer line breaks.

  Details of the containment pressure temperature transient analysis, and the subcompartmental differential pressure analyses are presented in FSAR Section 6.2.1. The structural details are described in FSAR Section 3.8. See FSAR Section 6.2.2 for heat removal; FSAR Section 3.8 for access openings; and FSAR Sections 3.8, 6.2.4, and 8.3 for penetrations.

- GDC-38, Containment Heat Removal, is described in FSAR Section 3.1.4.9.

  Two separate containment spray recirculation trains described in FSAR Section 6.2.2, each with 100-percent capacity, to remove heat from the Containment Building following certain design basis accidents. Each train contains two separate pumps, one heat exchanger, and seven spray headers, and each system is fed from its individual electrical Class 1E bus. Each Class 1E bus is connected to a separate offsite power source and is also connected to its individual onsite power source as described in FSAR Section 8.3. Containment isolation valves separate all components from the containment penetrations.

- GDC-13, Instrumentation and Control, is described in FSAR Section 3.1.2.4.

  To ensure adequate safety, instrumentation and control systems are provided to monitor and control significant variables over their anticipated range for all conditions in the reactor core, RCS, steam and power conversion system, radioactive waste systems, ESF systems, and the Containment Building. Parameters that must be provided for operator use under normal operating and accident conditions are indicated in the Control Room in close proximity to the controls which maintain the indicated parameters in the proper range.

  The quantity and types of process instrumentation provided ensures safe and orderly operation of all systems over the full design range of the plant. These systems are described in FSAR Chapters 6 through 12.

- GDC-64, Monitoring Radioactivity Releases, is described in FSAR Section 3.1.6.5.

  The Containment Building atmosphere is monitored continuously during normal and transient operations, using particulate, iodine and gaseous monitors. Under post-accident conditions, samples of the containment atmosphere provide data of existing airborne radioactive concentrations within the Containment Building. Radioactivity levels contained in the facility effluent discharge paths and in the plant environs are continuously monitored during normal and accident conditions by the Process Radiation Monitoring System described in FSAR Section 11.5. Provisions for monitoring the plant areas for radioactivity are included in the area radiation monitoring system described in FSAR Section 12.3.4. In addition to the installed detectors, periodic surveys are conducted using portable equipment, as discussed in FSAR Section 12.5.

2.6.1.2 Technical Evaluation

2.6.1.2.1 Introduction

2.6.1.2.1.1 Loss-of-Coolant Accident

The evaluation of the design basis LOCA event relative containment peak pressure and temperature response was completed to demonstrate the acceptability of the containment heat removal system to mitigate the consequences of a LOCA inside containment and to support the stretch power uprate (SPU) program operation. This evaluation is documented in the subsections below.

The containment response analysis demonstrates the acceptability of the containment heat removal systems to mitigate the consequence of a large LOCA inside containment. The impact of LOCA mass and energy (M&E) releases on the containment pressure and temperature are addressed to assure that the containment pressure and temperature remain below their respective design limits. The systems must also be capable of maintaining the equipment qualification (EQ) parameters to within acceptable limits at the SPU program conditions. The containment design pressure for CPNPP is 50 psig. The containment design temperature is 280°F.

The CPNPP LOCA containment response analysis considered a spectrum of cases as discussed in Licensing Report (LR) subsection 2.6.3.1, Mass and Energy Release for Postulated Loss-of-Coolant Accidents. The cases address break location, and postulated single failure (minimum and maximum safeguards). Only the limiting cases, which address the containment peak pressure case and limiting long-term EQ case, are presented herein. The LOCA pressure and temperature response analyses were performed assuming a loss-of-offsite power and a worst single failure (loss of one emergency diesel generator (EDG) or the loss of one containment spray train).

Calculation of the containment response following a postulated LOCA was analyzed by use of the digital computer code GOTHIC. GOTHIC version 7.2a was used for the LOCA containment response analysis. The GOTHIC Technical Manual (Reference 1) provides a description of the governing equations, constitutive models, and solution methods in the solver. The GOTHIC Qualifications Report (Reference 2) provides a comparison of the solver results with both analytical solutions and experimental data.

The GOTHIC containment modeling for CPNPP is consistent with the recent NRC approved Ginna evaluation model (Reference 3). The latest code version is used to take advantage of the diffusion layer model (DLM) heat transfer option. This heat transfer option was approved by the NRC (Reference 3) for use in Ginna containment analyses with the condition that mist be excluded from what was earlier termed as the mist diffusion layer model (MDLM). The GOTHIC containment modeling for CPNPP has followed the conditions of acceptance placed on Ginna. The differences in GOTHIC code versions are documented in Appendix A of the GOTHIC User Manual Release Notes (Reference 4). Version 7.2a is used consistent with the restrictions identified in Reference 3; none of the user-controlled enhancements added to version 7.2a were

implemented in the CPNPP containment model. A description of the CPNPP GOTHIC model is provided in LR subsection 2.6.1.2.3 - Description of Analyses and Evaluations.

The containment response for design basis LOCA and steam line break containment integrity is an American Nuclear Society (ANS) Condition IV event, an infrequent fault. The relevant requirements to satisfy NRC acceptance criteria are as follows:

- GDC-16 and -50: In order to satisfy the requirement of GDC-16 and -50, the peak calculated containment pressure should be less than the containment design pressure of 50 psig.

- GDC-38: In order to satisfy the requirement of GDC-38, the calculated pressure at 24 hours should be less than 50 percent of the peak calculated value. (This is related to the criteria for containment leakage assumptions as affecting doses at 24 hours.)

2.6.1.2.1.2 Main Steam Line Break

Steam line ruptures occurring inside the reactor containment may result in significant releases of high-energy fluid to the containment environment that could produce high-pressure conditions for extended periods of time. The magnitude of the releases following a steam line rupture is dependent upon the plant initial operating conditions and the size of the rupture as well as the configuration of the plant steam system and containment design. Because of these effects, the steam line break and containment response analysis considers a spectrum of cases that vary the initial condition, break size, and postulated single failure. The M&E release analysis was reanalyzed at the SPU conditions, and a description of this reanalysis, including the mass and energy results, is provided in LR subsection 2.6.3.2, Mass and Energy Release for Postulated Secondary-System Pipe Ruptures.

The containment integrity analysis (also performed with the GOTHIC computer code as described in LR subsection 2.6.1.2.1.1) was also performed to confirm that the pressure and temperature inside containment will remain below the pressure and temperature limits for a postulated system pipe rupture. The system must also be capable of maintaining the EQ parameters to within acceptable limits at the uprate program conditions.

2.6.1.2.2 Input Parameters, Assumptions, and Acceptance Criteria

The major modeling input parameters and assumptions that are used in the CPNPP containment evaluation model for the LOCA and steam line break events are identified in this section. The assumed initial conditions and input assumptions associated with the containment sprays are listed in Table 2.6.1-1. The primary function of the residual heat removal system (RHRS) is to remove heat from the core by way of the ECCS. The recirculation system alignment is outlined in Table 2.6.1-2. The containment structural heat sink input is provided in Table 2.6.1-3, and the corresponding material properties are listed in Table 2.6.1-4.

The LOCA containment analysis described herein utilized inputs and assumptions in support the SPU program, while addressing analytical conservatisms. In changing from the CONTEMPT

code to the GOTHIC code, the known difference that exists between the current licensing analysis and the SPU program containment integrity analysis is the treatment of the containment spray flow. The efficiency of the containment spray system was modeled in GOTHIC as 84.2 percent during the injection phase and the recirculation phase. Since the remaining spray flow was not credited for atmospheric heat removal, it was also not credited for any heat removal from any of the containment structures or the containment sump. In the current licensing basis analysis, 15.8 percent of the spray flow does not cool the atmosphere or the heat sinks but it does contribute to cooling the liquid in the sump.

The CPNPP containment design pressure is 50.0 psig and the design temperature is 280°F. The containment pressure at 24 hours must be less than 50 percent of the calculated peak containment pressure for the LOCA containment analysis. The containment steam temperature transients must be less than the equipment qualification (EQ) profile.

2.6.1.2.3 Description of Analyses and Evaluations

Noding Structure

The CPNPP GOTHIC containment evaluation model for the LOCA and steam line break events consisted of one volume. Additional boundary conditions, volumes, flow paths, and components are used to model accumulator nitrogen release and sump recirculation. Injection of accumulator nitrogen during a LOCA event is modeled by a boundary condition. The recirculation system model uses GOTHIC component models for the RHR and component cooling water (CCW) heat exchangers (HXs) and the CCW pumps. Recirculation flow from the sump is modeled using a boundary condition.

Volume Input

GOTHIC requires the volume, height, diameter, and elevation input values for each node. The containment is modeled as a single control volume in the containment model. The minimum free volume of 2,985,000 ft^3 was used.

Initial Conditions

The containment initial conditions are listed below:

- Pressure: 16.2 psia
- Relative Humidity: 15 percent
- Temperature: 120°F

Flow Paths

Flow paths connect the boundary conditions to the containment volume. The flow rate is specified by the boundary condition. Standard values are used for the area, hydraulic diameter, friction length, and inertia length of the flow path. Since this is a single lumped parameter

volume model, the elevation of the break flow paths is arbitrarily set to 50 feet and the elevation of the spray flow paths is arbitrarily set to 90 feet.

Flow boundary conditions model the break flow to the containment. The boundary conditions are linked to functions that define the M&E of the break flow. The boundary conditions are connected to the containment control volume via flow paths.

The containment spray is also modeled as a boundary condition. It is connected to the containment control volume via a flow path.

Heat Sinks

The heat sinks in the containment are modeled as GOTHIC thermal conductor. The heat sink data is based on conservatively low surface areas and is summarized in Table 2.6.1-3.

A thin air gap is assumed to exist between the steel and concrete for steel-jacketed heat sinks. A gap conductance of 0.0161 Btu/hr/ft^2/°F is assumed between steel and concrete. The gap width is determined by dividing the gap thermal conductivity by the gap conductance.

The specific heat and thermal conductivity for the heat sink materials is summarized in Table 2.6.1-4. The specific heat value was calculated based on the volumetric heat capacity.

Heat and Mass Transfer Correlations

GOTHIC has a number of heat transfer coefficient options that can be used for containment analyses.

The direct heat transfer coefficient set is used, along with the DLM mass transfer correlation, for all of the heat sinks inside containment. This heat transfer methodology was reviewed and approved for use in the Ginna containment design basis accident analyses (Reference 3). The DLM correlation does not require the user to specify a revaporization input value.

The direct heat transfer coefficient set is used for the heat sinks representing floors. The submerged conductors are essentially insulated for the vapor after the pool develops. Insulated surfaces are modeled with no heat loss (0.0 Btu/hr-ft^2/°F).

Modeling Sump Recirculation

A sump recirculation model consisting of simplified RHRS and CCW system models was included in the CPNPP containment model to calculate the long-term LOCA containment pressure and temperature response. The RHR heat exchanger cools the water from the containment sump. The RHR system injects the cooled water into the RCS to cool the core. The RHR heat exchanger is cooled with CCW water and service water provides the ultimate heat sink, cooling the CCW heat exchangers.

Boundary Conditions

LOCA Mass and Energy Release

The LOCA mass and energy release methodology generates the releases from both sides of the break (or two flow paths: M&E exiting from the vessel side of the break; and M&E exiting from the steam generator side of the break – as defined for a double-ended hot leg break). The LOCA transient M&E releases are calculated as separate flow paths (for the first 3,600 seconds) and input to the GOTHIC containment model via flow boundary conditions. The flow boundary conditions are linked to functions that define the mass break flow and the enthalpy of the break flow. The break mass and enthalpy are input to the containment model as external functions defined by control variables. The M&E releases from the boundary conditions are analyzed for CPNPP out to 3,600 seconds; that is, the time at which all energy in the primary heat structures and steam generator secondary system is released/depressurized to atmospheric pressure (14.7 psia and 212°F). LR subsection 2.6.3.1, M&E Release Analysis for Postulated Loss-of-Coolant Accidents, describes the LOCA long-term M&E release methodology (Reference 6).

The liquid portion of the break flow is released as drops with an assumed diameter of 100 microns (0.00394 in). This is consistent with the methodology approved for Ginna (Reference 3) and is based on data presented in Reference 5.

The long-term, post 3,600 second, mass and release (boil-off from the core at the decay heating rate) calculations are performed through user defined functions by GOTHIC. These input functions are used to incorporate the sump water cooling in the long term and are consistent with the Westinghouse methodology previously approved by the NRC. After primary system and secondary system energy have been released (depressurized to atmospheric pressure, (14.7 psia and 212°F), the M&E release to the containment is assumed to be from long-term steaming of decay heat. A flow boundary condition is defined to provide the long-term boil-off M&E release to containment. The mass flow rate and enthalpy of the flow is calculated using GOTHIC control variables.

The ANS Standard 5.1 (Reference 9) decay heat model (+2σ uncertainty) is used to calculate the long-term boil-off from the core. Table 2.6.3.1-4 lists the decay heat curve used. All of the decay heat is assumed to produce steam from the recirculated ECCS water. The remainder of the ECCS water is returned to the sump region of the containment control volume. These assumptions are consistent with the long-term M&E methodology documented in Reference 6.

Steam Line Break Mass and Energy Release

The transient steam line break mass and energy releases are calculated and input into the GOTHIC containment model using a single boundary condition. The M&E releases are analyzed out to approximately 650 seconds, which is when there are no more M&E releases from the faulted steam generator. The mass and energy release methodology utilizes the RETRAN code, documented in WCAP-14882 (Reference 7). Additional information including a detailed discussion of the event is provided in LR subsection 2.6.3.2.

Containment Spray System

The containment spray is modeled with a boundary condition. CPNPP has two trains of containment safeguards available, with two spray pumps per train.

Consistent with the application of single-failure criterion presented in LR subsection 2.6.3.1.2.1.2, Application of Single-Failure Criterion, an inherent assumption is that offsite power is lost with the pipe rupture for the LOCA event. This results in the actuation of the EDG, powering the two trains of safeguards equipment. Operation of the EDG delays the operation of the safeguards equipment that is required to mitigate the transient. Relative to single failure criterion with respect to a LOCA event, one spray train is considered inoperable, whether due to a EDG failure (minimum safeguards case) or as the limiting single failure in the maximum safeguards case.

For the steam line break event, one of the assumed single failures is the containment safeguards failure. This is the loss of one safeguards train, which reduces the active heat removal. If the limiting single failure is in a secondary system, both trains of containment spray are operational for mitigation of the containment transient. It is assumed that the steam line break scenarios have offsite power available. Therefore, the diesel generators do not factor into the plant response.

The containment spray is modeled to actuate on the containment spray pressure setpoint with a biased high uncertainty and to begin injecting 120°F water. The setpoint and associated delay is designed to conservatively maximize the time it takes for the sprays to start injecting. The LOCA and large steam line break containment response analyses use the larger time delay of 74.3 seconds after the containment reaches the Hi-1 setpoint (5.0 psig) or 52.5 seconds after the containment pressure reaches the Hi-3 setpoint (20.0 psig). The containment spray flow is 4,835.61 gpm per spray train in the injection phase considering the 84.2-percent efficiency. The spray flow rate is modeled in GOTHIC as a function of time. The containment spray is credited during the injection phase of the transient and during the cold leg and hot leg recirculation phases of the transient as the remaining spray train is the only active heat removal system. The containment spray flow is 5,784.54 gpm per spray train during the recirculation phase considering the 84.2-percent efficiency.

Accumulator Nitrogen Gas Modeling

The accumulator nitrogen gas release is modeled with a flow boundary condition in the LOCA containment model. The nitrogen release rate was conservatively calculated by maximizing the mass available to be injected. The nitrogen gas release rate was used as input for the GOTHIC function, as a specified rate over a fixed time period. Nitrogen gas was released at a rate of 348.8 lbm/seconds, beginning at 45.44 seconds (average accumulator tank water volume empty time) and ending at 65.64 seconds.

2.6.1.2.4 Primary Containment Functional Design Results

2.6.1.2.4.1 Loss-of-Coolant Accident

The containment pressure and steam temperature profiles for the DEHL case (peak pressure case) are shown in Figures 2.6.1-1 and 2.6.1-2. Table 2.6.1-5 provides the transient sequence of events for the double ended hot leg (DEHL) transient. The results of the double-ended pump suction (DEPS) (long-term EQ transient) are shown in Figures 2.6.1-3 through 2.6.1-5. Table 2.6.1-6 presents sequence of events for the DEPS transient. Table 2.6.1-7 provides the containment pressure and temperature results relative to peak containment conditions and also at 24 hours for EQ support and the acceptance limits for these parameters.

A review of the results presented in Table 2.6.1-7, shows that the analysis margin (analysis margin is the difference between the calculated peak pressure and temperature and the acceptance limits) is maintained. The current licensing containment response basis results for containment peak pressure and temperature for a LOCA event was 44.9 psig and 272.1°F, respectively. From the containment response analysis, performed in support of the CPNPP SPU, the containment peak pressure and temperature is 44.45 psig and 266.1°F.

Refer to LR Section 2.3.1 for impact on the equipment qualification.

LOCA Containment Response Transient Description Double Ended Pump Suction Break with Minimum Safeguards

This analysis assumes a loss-of-offsite power coincidence with a double-ended rupture of the RCS piping between the steam generator outlet and the RCS pump inlet (suction). The associated single-failure assumption is the failure of a diesel generator to start resulting in one train of ECCS and containment safeguards equipment being available. The containment heat removal systems that are assumed available are one RHR heat exchanger, one CCW heat exchanger, and one containment spray train. Further, loss-of-offsite power delays the actuation times of the safeguards equipment due to the time required for diesel startup after receipt of the safety injection signal.

The postulated RCS break results in a rapid release of M&E to the containment with a resultant rapid increase in both the containment pressure and temperature. This rapid rise in containment pressure actuates the containment spray actuation pressure signal at 0.65 seconds. The containment pressure continues to rise rapidly in response to the release of M&E, reaching the blowdown peak pressure of 40.01 psig at 24.01 seconds, and then decreasing slightly as the end of blowdown occurs at 26.0 seconds (pressure of 39.92 psig). The end of blowdown marks a time when the initial inventory in the RCS has been exhausted and a slow process of filling the RCS downcomer in preparation for reflood has begun. Since the M&E release during this period is low, pressure continues to decrease slightly. At approximately 46.1 seconds the accumulators have emptied, and the pressure increases in response to the loss of steam condensation in the RCS loops and the introduction of the accumulator nitrogen gas to containment out to a second peak which occurred at 68.56 seconds.

During this period, the containment spray (74.95 seconds) has also started and is removing heat. Reflood continues at a reduced flooding rate due to the buildup of mass in the RCS core, which offsets the downcomer head. This reduction in flooding rate and the containment injection spray leads to a slowly decreasing containment pressure out to the end of reflood, which occurs at 198.9 seconds.

At this juncture, by design of WCAP-10325 (Reference 6) mass and energy release evaluation model, energy removal from the steam generator secondary side begins at a very high rate, resulting in a rise in containment pressure from 198.9 seconds out to 446.6 seconds where another pressure peak of 42.95 psig is reached. Energy continues to be removed from the secondary side of the broken loop and intact loop steam generators until the secondary temperature is the saturation temperature (T_{sat}) at the containment design pressure. This point is reached at 449.0 and 1,572.4 seconds for the broken loop and intact loop steam generators, respectively. Energy removal from the secondary side of the steam generators continues by way of intermediate pressure equilibration stages until the final depressurization, when the secondary reaches the mandatory reference temperature of T_{sat} at 14.7 psia, and 212°F, at 3,600 seconds. The heat removal of the broken loop and intact loop steam generators are calculated separately. The intermediate equilibration stages are met at 586.8 seconds for the broken loop steam generator and 1,726.8 seconds for the intact loop steam generator. After the peak containment pressure is reached and during the steam generator depressurization period, the M&E release is reduced since the large energy removal has been accomplished. Containment pressure slowly decreases through the initiation of cold leg recirculation at 1,159.8 seconds. At this time, the ECCS is realigned for sump recirculation resulting in an increase in safety injection temperature (due to the delivery from the hot sump and a reduction in steam condensation). Also at 1,721.7 seconds the containment injection phase spray is terminated from the refueling water storage tank. At this point another intermediate pressure peak occurs of 44.44 psig. The ultimate peak pressure of 44.45 psig is reached during this period (at 3,407 seconds). At this time, the energy removal from the recirculation spray flow exceeds the energy release and the pressure and temperature turn around. This trend continues to the end of the transient.

The LOCA containment response analysis has been performed as part of the SPU for CPNPP. As illustrated in the LR subsection 2.6.1.2.4, Primary Containment Functional Design Results, all cases were well below the containment acceptance limits of 50 psig (64.7 psia) and 286°F. In addition, the long term DEPS case was well below 50 percent of the peak containment pressure value within 24 hours. Based on the results, all applicable criteria have been met.

2.6.1.2.4.2 Main Steam Line Break

Sixteen steam line break cases were analyzed varying the initial reactor power and the assumed single failure. The M&E release methodology utilizes the RETRAN code, documented in WCAP-14882 (Reference 7). The analysis included the effects of the uprate to 3,628 MWt NSSS power and a shutdown margin of 1.3 percent. Eight cases consider a double-ended rupture immediately downstream of the flow restrictor at the outlet of the steam generator. This conservatively bounds the plant response to any smaller break. This break location also limits the break size to the 1.4 ft^2 cross-sectional area of the flow restrictor. The remaining eight

cases model a split rupture, the largest break that will neither generate a steam line isolation signal nor result in water entrainment in the break effluent as outlined in Reference 8. Additional information including a description of the assumed single failures is provided in LR subsection 2.6.3.2.

As mentioned previously, the steam line break containment response analysis uses the GOTHIC, version 7.2a, computer code. The GOTHIC Technical Manual (Reference 1) provides a description of the governing equations, constitutive models, and solution methods in the solver. The GOTHIC Qualifications Report (Reference 2) provides a comparison of the solver results with both analytical solutions and experimental data.

The peak pressures and peak temperatures from the spectrum of cases are listed in Table 2.6.1-8. A review of the results shows that analysis margin is maintained. The current licensing containment response basis results for containment peak pressure and temperature for a steam line break event was 46.5 psia and 343.5°F, respectively. From the containment response analysis, performed in support of the SPU, the containment peak pressure and temperature is 39.0 psig and 324.9°F.

The limiting peak pressure case presented herein is a 30-percent power, 4.7 ft^2 split break assuming a containment safeguards failure. The sequence of events for this limiting case is listed in Table 2.6.1-9. Plots of the mass and enthalpy for this case are provided in LR subsection 2.6.3.2, and the containment pressure transient for the limiting case is in Figure 2.6.1-6. The peak containment pressure is 39.0 psig which is less than the limit of 50.0 psig. Therefore, the acceptance criterion is met.

A temperature composite has been developed for equipment qualification purposes inside containment. The composite was created by determining the maximum GOTHIC temperature for the sixteen analyzed uprate cases at each time step in the transient. A plot of the temperature composite is provided in Figure 2.6.1-7.

2.6.1.3 Conclusion

Luminant Power has reviewed the assessment of the containment pressure and temperature transient and concludes that it has adequately accounted for the increase of M&E releases that would result from the proposed uprate. Luminant Power further concludes that containment systems will continue to provide sufficient pressure and temperature mitigation capability to ensure that containment integrity is maintained. Luminant Power also concludes that the containment systems and instrumentation will continue to be adequate for monitoring containment parameters and release of radioactivity during normal and accident conditions and will meet the current licensing basis requirements with respect to GDCs -13, -16, -38, -50, and -64 following implementation of the proposed uprate.

2.6.1.4 References

1. NAI 8907-06, Rev. 16, "GOTHIC Containment Analysis Package Technical Manual," Version 7.2a, January 2006.

2. NAI-8907-09, Rev. 9, "GOTHIC Containment Analysis Package Qualification Report," Version 7.2a, January 2006.

3. Docket No. 50-244, Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 97, to Renewed Facility Operating License No. DPR-18 R. E. Ginna Nuclear Power Plant, Nuclear Regulatory Commission, July 11, 2006

4. NAI 8907-02, Rev. 17, "GOTHIC Containment Analysis Package User Manual," Version 7.2a, January 2006.

5. AICHE Journal Volume 8, #2, "Sprays formed by Flashing Liquid Jets," May 1962.

6. WCAP-10325, May 1983 and WCAP-10326, "Westinghouse LOCA Mass and Energy Release Model for Containment Design March 1979 Version."

7. WCAP-14882, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," April 1999.

8. WCAP-8822 and WCAP-8860, "Mass and Energy Releases Following a Steam Line Rupture," September 1976; WCAP-8822 and WCAP-8860, "Supplement 1 – Calculations of Steam Superheat in Mass/Energy Releases Following a Steam Line Rupture," September 1986; WCAP-8822 and WCAP-8860, "Supplement 2 – Impact of Steam Superheat in Mass/Energy Releases Following a Steam Line Rupture for Dry and Subatmospheric Containment Designs," September 1986.

9. ANSI/ANS-5.1 1979, American National Standard for Decay Heat Power in Light Water Reactors, August 29, 1979.

Table 2.6.1-1 Containment Response Analysis Parameters	
Parameter	**Value**
Service Water Temperature (°F)	102.0
RWST Water Temperature (°F)	120
Initial Containment Temperature (°F)	120
Initial Containment Pressure (psig)	1.5
Initial Relative Humidity (%)	15.0
Net Free Volume (ft^3)	2,985,000
Containment Spray Trains	
Total (there are two pumps per train)	2
Analysis Diesel Failure or Single Containment Safeguards Failure	1
Analysis No Failure	2
Flowrate (gpm) Injection Phase (per train - with efficiency adjustment)	4,835.61
Flowrate (gpm) Recirculation Phase (per train - with efficiency adjustment)	5,784.54
Spray System Efficiency (%)	84.2
Containment Spray Actuation Setpoint (psig)	5.0/20.0[(1)]
Delay Time (sec)	74.3/52.5[(1)]
ECCS Recirculation Switchover, (sec) Minimum Safeguards Maximum Safeguards	 1,159.81 941.55
Injection Spray Termination Time, (sec) Minimum Safeguards Maximum Safeguards	 1,721.69 1,397.69
Containment ECCS Sump Cold Leg Recirculation Flow, (gpm) Minimum Safeguards Maximum Safeguards	 3,433.5 8,546.0
Containment ECCS Sump Hot Leg Recirculation Flow, (gpm) Minimum Safeguards Maximum Safeguards	 3,057.5 5,333.1

Note:

1. The LOCA and steam line break containment response analyses use the larger time of 74.3 seconds after the containment reaches the Hi-1 setpoint (5.0 psig) or 52.5 seconds after the containment pressure reaches the Hi-3 setpoint (20.0 psig).

Table 2.6.1-2

LOCA Containment Response Analysis Recirculation System Alignment Parameters

Residual Heat Removal System	
RHR Heat Exchangers	
Maximum number	2
Modeled in analysis[1]	1
Recirculation switchover time, sec	
Minimum safeguards	1,159.81
Flow rate, lbm/s	
Tube side	527.8
Shell side	1,055.6
Component Cooling Water Heat Exchangers	
Maximum number	2
Modeled in analysis	1
Flowrate, lbm/s	
Shell side[1]	2,044.6
Tube side[1] (service water)	1,947.3
Additional heat loads, Btu/hr	12,492.3

Note:

1. Minimum heat removal data representing 1 EDG

Table 2.6.1-3

Containment Structural Heat Sink Input

Heat Sink Number	Description	Area (ft^2)	Material	Thickness (inches)
1	Containment dome hemisphere	28,628	Paint on steel	0.006996
			Steel	0.500400
			Liner air gap	0.124800
			Concrete	29.499600
			Paint on concrete	0.027996
2	Misc. steel (filters, coils, hangers)	23,819	Paint on steel	0.007
			Steel	0.263
3	Misc. steel (piping, shielding, beams)	4,586	Paint on steel	0.006996
			Steel	0.160980
4	Slabs and walls, 2-sided	172,676	Paint on concrete	0.027996
			Concrete	12.000000
5	Ventilation ducts	40,257	Paint on steel	0.006996
			Steel	0.060000
6	Misc. steel (platforms, ladders, ...)	6,334	Paint on steel	0.006996
			Steel	0.049992
7	Cylindrical section of containment wall	81,642	Paint on steel	0.006996
			Steel	0.375600
			Liner air gap	0.124800
			Concrete	53.624400
			Paint on concrete	0.027996
8	Misc. steel (crane wheel, beams, ...)	1,866	Paint on steel	0.006996
			Steel	0.750000
9	Misc. steel (wheels, girders, ...)	528	Paint on steel	0.006996
			Steel	4.002000

Table 2.6.1-3 (cont.) Containment Structural Heat Sink Input				
Heat Sink Number	**Description**	**Area (ft^2)**	**Material**	**Thickness (inches)**
10	Foundation mat	11,395	Paint on concrete	0.027996
			Concrete	30.000000
			Liner air gap	0.124800
			Steel	0.240000
			Liner air gap	0.124800
			Concrete	143.964000
11	Misc. steel (boxes, pumps, ...)	23,359	Paint on steel	0.006996
			Steel	0.150000
12	Misc. steel (piping, rims, crane, ...)	21,684	Paint on steel	0.006996
			Steel	0.279600
13	Misc. steel (trolley girders, ...)	1,528	Paint on steel	0.006996
			Steel	2.000400
14	Girders	16,631	Paint on steel	0.006996
			Steel	2.250000
15	Trolley girders	22,987	Paint on steel	0.006996
			Steel	0.096000

Table 2.6.1-4 Material Properties for Containment Structural Heat Sink		
Material	**Conductivity (Btu/hr-ft-°F)**	**Specific Heat (Btu/lbm-°F)**
Steel	26	0.11
Concrete	0.8	0.16
Paint on Steel	0.0875	0.1
Paint on Concrete	0.07	0.1
Liner Air Gap	0.0161	0.20875

Table 2.6.1-5 Double-Ended Hot Leg Break Sequence of Events	
Time (sec)	**Event Description**
0.0	Break Occurs , and Loss-of-Offsite Power are Assumed
2.2	Compensated Pressurizer Pressure for Reactor Trip (1,845 psig) Reached and Turbine Trip Occurs
3.9	Low-Pressurizer Pressure SI Setpoint (1,700.3 psig) Reached – Feedwater Isolation Signal
10.91	Feedwater Isolation Valves Closed
12.4	Broken Loop Accumulator Begins Injecting Water
12.4	Intact Loop Accumulator Begins Injecting Water
21.5	Peak Temperature Occurs (260.5°F)
22.0	Peak Pressure Occurs (40.83 psig)
22.4	End of Blowdown Phase
50.0	Transient Modeling Terminated

Table 2.6.1-6 Double-Ended Pump Suction Break Sequence of Events (Minimum Safeguards)	
Time (sec)	**Event Description**
0.0	Break Occurs and Loss of Offsite Power is Assumed
.65	Containment Spray Actuation Pressure Setpoint (5.0 psig; Analysis Value) Reached
2.6	Compensated Pressurizer Pressure Reactor Trip (1,845 psig) Reached and Turbine Trip Occurs
4.3	Low Pressurizer Pressure SI Setpoint (1,700.3 psig) Reached (Safety Injection Begins coincident with Low Pressurizer Pressure SI Setpoint)
11.31	Feedwater Isolation Valves Closed
15.1	Broken Loop Accumulator Begins Injecting Water
15.3	Intact Loop Accumulator Begins Injecting Water
26.0	End of Blowdown Phase
26.0	Accumulator Mass Adjustment for Refill Period
31.30	Pumped Safety Injection Begins (Includes 27 Second Diesel Delay)
44.76	Broken Loop Accumulator Water Injection Ends
46.11	Intact Loop Accumulator Water Injection Ends
74.95	Containment Spray Pump (RWST) Begins
198.9	End of Reflood for Minimum Safeguards Case
446.6	Containment Peak Temperature Occurs (266.1°F)
449.0	M&E Release Assumption: Broken Loop Steam Generator (SG) Equilibration when the Secondary Temperature is the Saturation (T_{sat}) At Containment Design Pressure of 50 psig
586.8	M&E Release Assumption: Broken Loop SG Equilibration at Containment Pressure of 40 psig
1,159.8	Switchover to Cold Leg Recirculation Begins
1,572.4	M&E Release Assumption: Intact Loop SG Equilibration when the Secondary Temperature is the Saturation (T_{sat}) at Containment Design Pressure of 50 psig
1,721.7	Containment Spray Terminated
1,726.8	M&E Release Assumption: Intact Loop SG Equilibration at Containment of 30 psig
3,407.0	Containment Peak Pressure Occurs (44.45 psig)
7.8E+6	Transient Modeling Terminated

Table 2.6.1-7 LOCA Containment Response Results				
Case	**Peak Press. @ Time**	**Peak Temp. @ Time**	**Peak Press. (psia) @ 24 hours**	**Peak Temp. (°F) @ 24 hours**
DEHL	40.83 psig @ 22.0 sec	260.5 °F @ 21.5 sec	NA	NA
DEPS-Minimum Safeguards	44.45 psig @ 3,407. sec	266.1 °F @ 446.6 sec	9.86	172.6
Containment Pressure – Acceptance Limits				
	Peak Pressure	**Pressure @ 24 hours**		
Pressure	50 psig	50% of the calculated peak pressure		
Containment Temperature – Acceptance Limits				
	Peak Temperature	**Temperature @ 24 hours**		
Temperature	280ºF	Less than EQ profile		

Table 2.6.1-8 Steam Line Break Containment Response Results				
Description			**Peak Press. (psig) @ Time (sec)**	**Peak Temp. (°F) @ Time (sec)**
Break	**Initial Power**	**Failure**		
1.4 ft^2 DER	100.6%	Cont. Safe. and FIV	32.7 @ 620	307.1 @ 80
1.4 ft^2 DER	70%	Cont. Safe. and FIV	34.3 @ 620	304.7 @ 80
1.4 ft^2 DER	30%	Cont. Safe. and FIV	36.6 @ 610	303.2 @ 90
1.4 ft^2 DER	0%	Cont. Safe. and FIV	36.4 @ 260	299.5 @ 100
4.3 ft^2 Split	100.6%	Cont. Safe. and FIV	34.2 @ 620	312.9 @ 70
4.5 ft^2 Split	70%	Cont. Safe. and FIV	35.9 @ 620	311.6 @ 60
4.7 ft^2 Split	30%	Cont. Safe. and FIV	39.0 @ 620	311.1 @ 50
4.7 ft^2 Split	0%	Cont. Safe. and FIV	38.3 @ 280	305.3 @ 50
1.4 ft^2 DER	100.6%	MSIV and FIV	33.0 @ 201	316.9 @ 51
1.4 ft^2 DER	70%	MSIV and FIV+	33.9 @ 240	313.8 @ 50
1.4 ft^2 DER	30%	MSIV and FIV	34.7 @ 360	312.0 @ 50
1.4 ft^2 DER	0%	MSIV and FIV	36.1 @ 260	305.9 @ 60
4.3 ft^2 Split	100.6%	MSIV and FIV	34.5 @ 200	324.9 @ 40
4.5 ft^2 Split	70%	MSIV and FIV	35.6 @ 240	324.3 @ 40
4.7 ft^2 Split	30%	MSIV and FIV	36.9 @ 380	324.1 @ 30
4.7 ft^2 Split	0%	MSIV and FIV	37.7 @ 280	318.2 @ 30

Notes:

FIV = feedwater isolation valve

MSIV = main steam isolation valve

Table 2.6.1-9

Steam Line Break Limiting Peak Containment Pressure Case Sequence of Events

Time (sec)	Event Description
2.4	Hi-1 Containment Pressure Setpoint Reached
4.0	Hi-1 Containment SI Setpoint Credited
4.0	AFW Initiation
4.6	Hi-2 Containment Pressure Setpoint Reached
6.0	Reactor Trip – Start of Rod Motion
11.0	Faulted Loop FIV/FCV Fully Closed
13.5	Faulted Loop MSIV Fully Closed
31.0	SI Flow Starts
76.7	Containment Sprays Start
94	SI Boron Reaches Core
620	Peak Containment Pressure Occurs
800	Mass Release Terminated

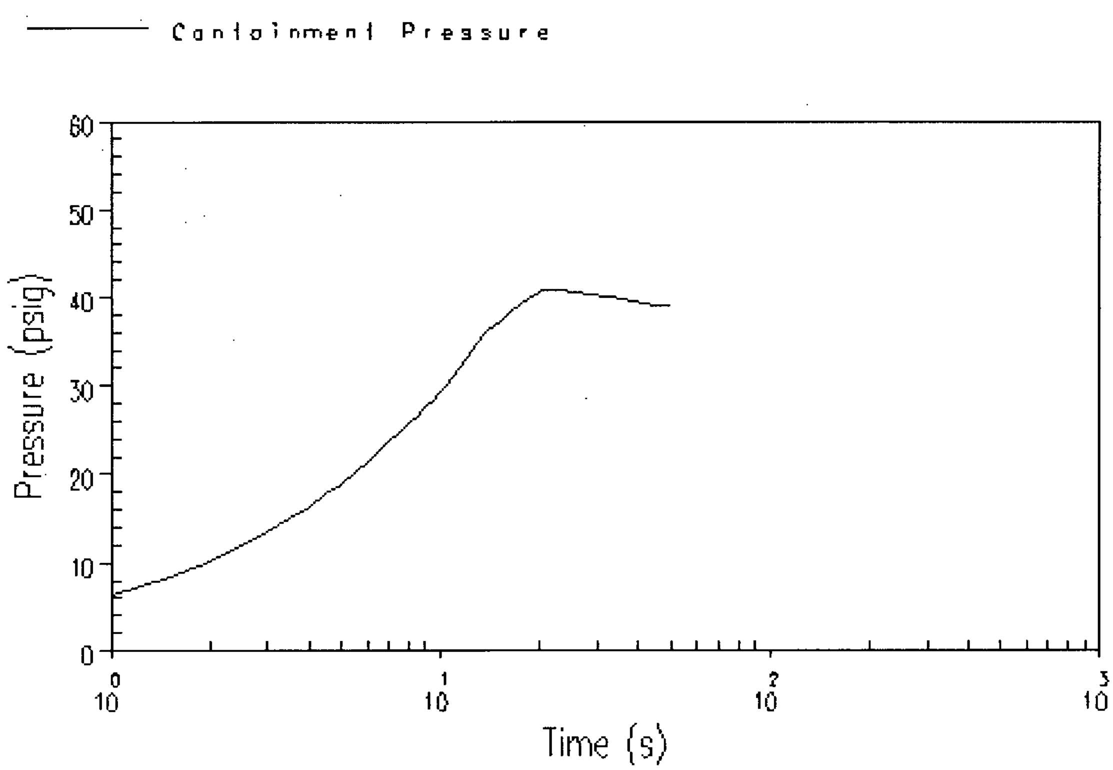

Figure 2.6.1-1 Containment Pressure – Double-Ended Hot Leg Break

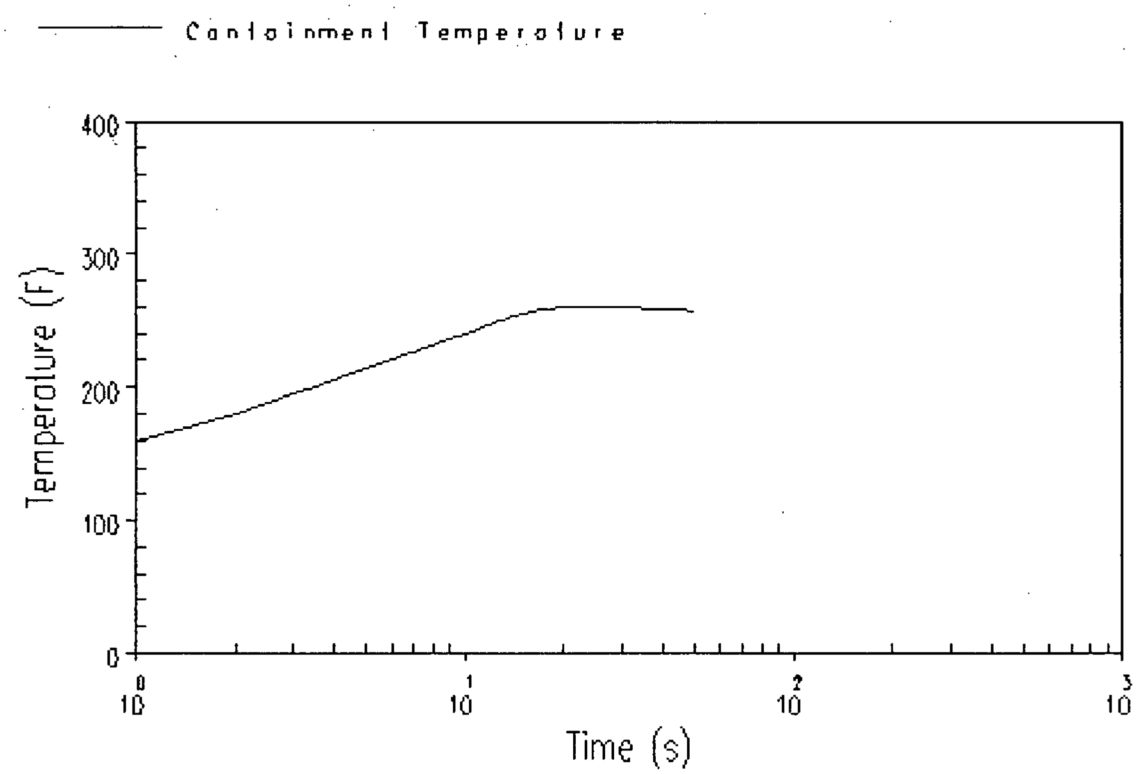

Figure 2.6.1-2 Containment Temperature – Double-Ended Hot Leg Break

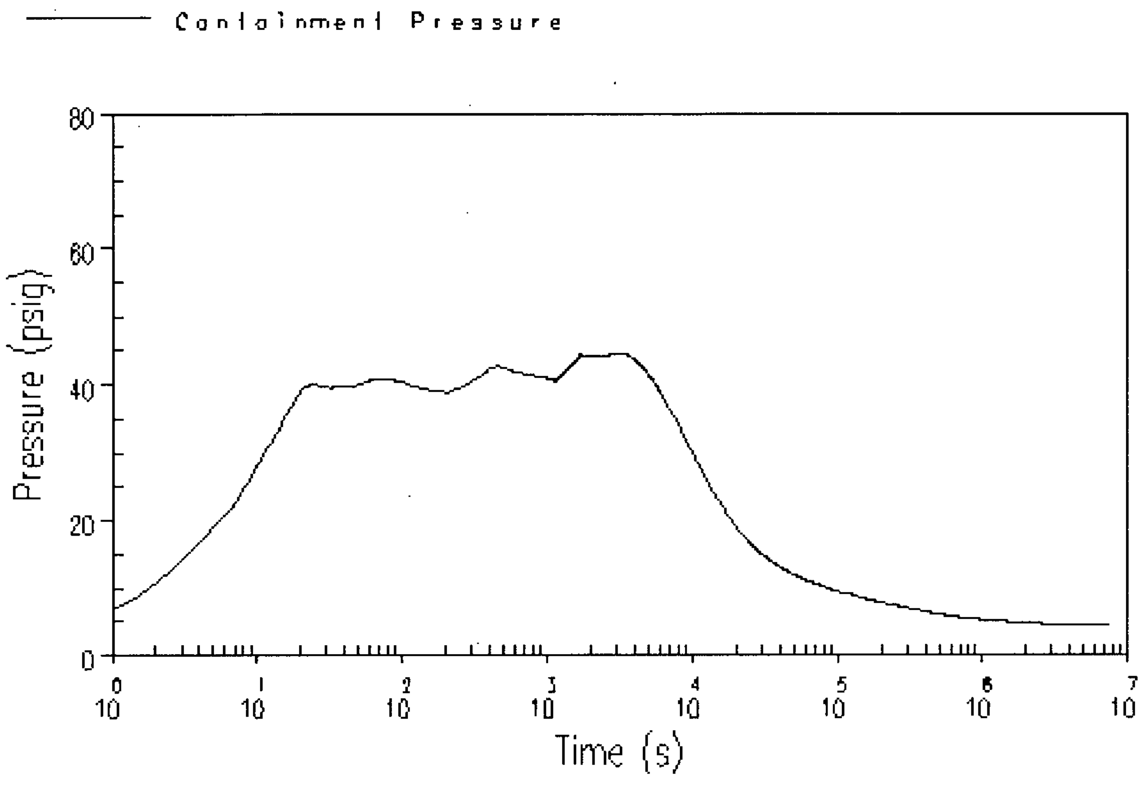

Figure 2.6-1-3 Containment Pressure – Double-Ended Pump Suction Break

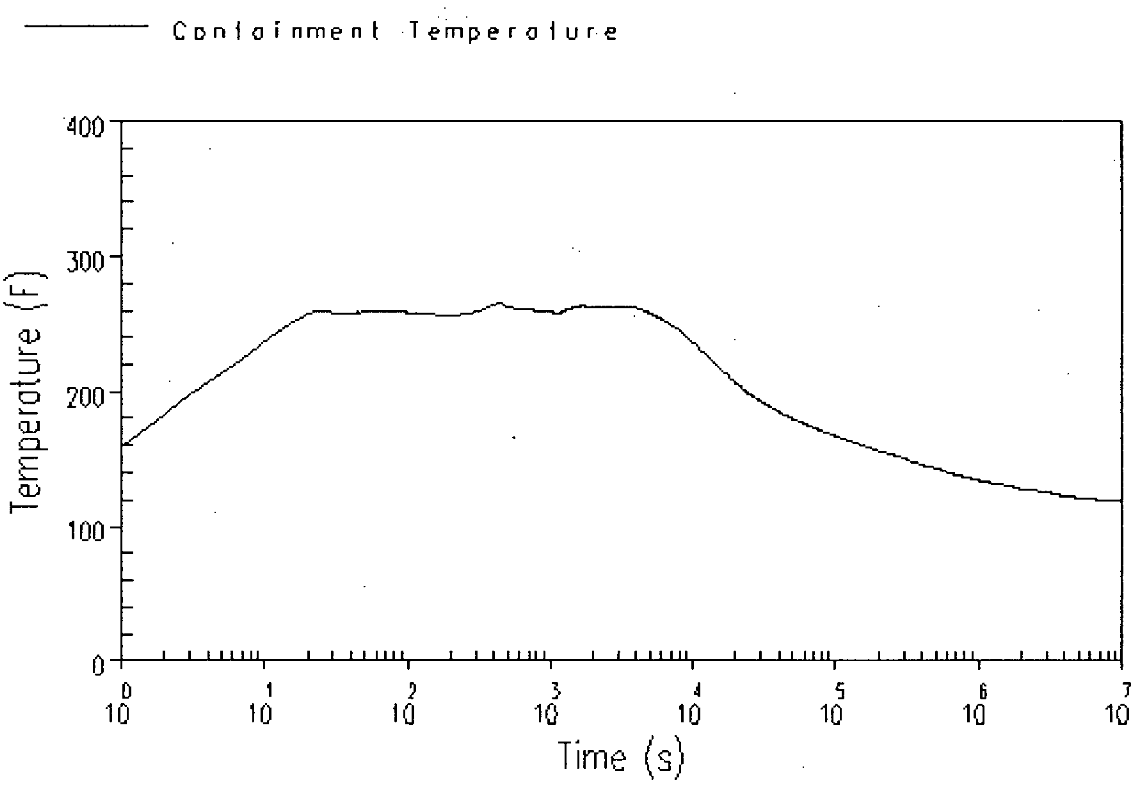

Figure 2.6.1-4 Containment Temperature – Double-Ended Pump Suction Break

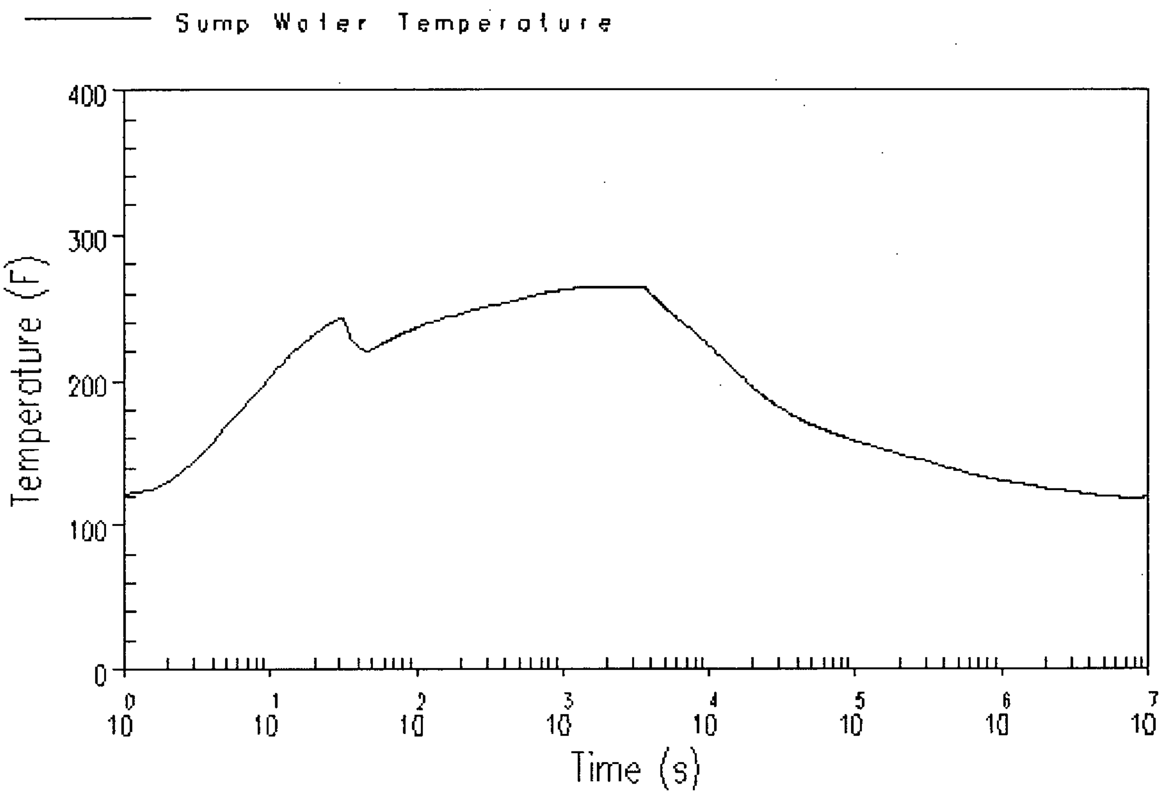

Figure 2.6.1-5 Containment Sump Temperature – Double-Ended Pump Suction Break

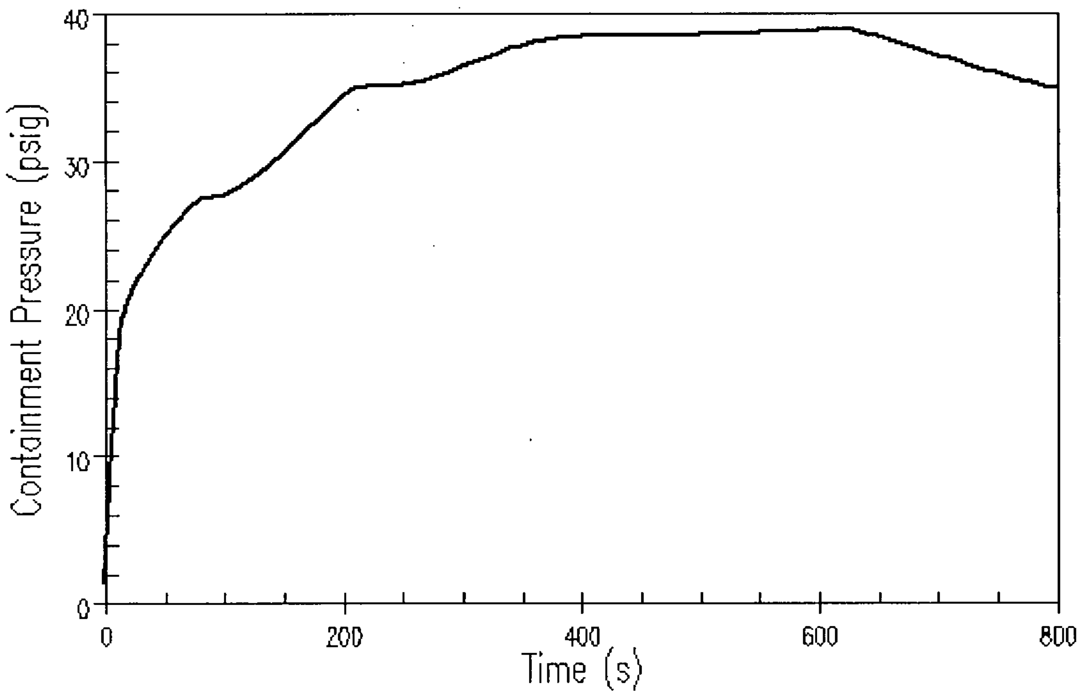

Figure 2.6.1-6 **Containment Pressure Response to a Steam Line Break – Peak Pressure Case 4.7 ft^2 Split Break, 30% Power, Containment Safeguards Failure**

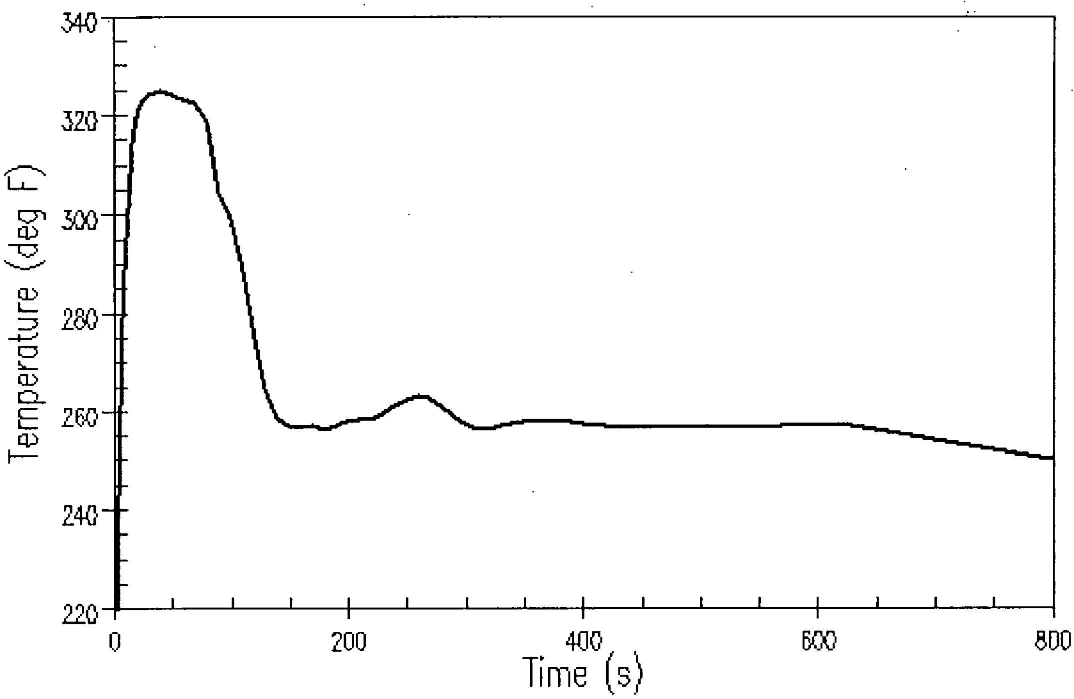

Figure 2.6.1-7 Containment Temperature Composite to a Steam Line Break

2.6.2 Subcompartment Analyses

2.6.2.1 Regulatory Evaluation

A subcompartment is defined as any fully or partially enclosed volume within the primary containment that houses high-energy piping and would limit the flow of fluid to the main containment volume in the event of a postulated pipe rupture within the volume. The Comanche Peak Nuclear Power Plant (CPNPP) review of the subcompartment analyses covered the determination of the design differential pressure values for containment subcompartments. The review focused on the effects of the increase in mass and energy (M&E) release into the containment due to operation at stretch power uprate (SPU) conditions and the resulting increase in pressurization.

The acceptance criteria for subcompartment analyses are based on:

- General Design Criterion (GDC)-4, insofar as it requires that structures, systems, and components (SSCs) important to safety be designed to accommodate the effects of, and be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, and that such SSCs be protected against dynamic effects.

- GDC-50, insofar as it requires that the containment subcompartments be designed with sufficient margin to prevent fracture of the structure due to the calculated pressure differential conditions across the walls of the subcompartments.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the adequacy of CPNPP design relative to conformance with:

- GDC-4 is described in the FSAR Section 3.1.1.4, Environmental and Dynamic Effect Bases.

  The station's SSCs important to safety are designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents (LOCAs). Environmental conditions are described in FSAR Section 3.11.

  These SSCs are appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that can result from equipment failures and from events and conditions outside the nuclear power unit.

Details of the design, environmental testing, and construction of these SSCs are included in FSAR Chapters 3, 5, 6, 7, 8, 9, and 10. Evaluation of the performance of safety features is contained in FSAR Chapter 15.

The leak-before-break (LBB) methodology demonstrates that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping. It has been applied to CPNPP Units 1 and 2 to exclude from the design basis the dynamic effects of postulated pipe ruptures in the primary coolant loop piping and 10-inch and larger reactor coolant loop branch lines, as discussed in FSAR Sections 3.6B.2.5.1 and 3.6B.2.5. Implementation of this technology eliminates the need for pipe whip restraints and jet impingement barriers, respectively. Containment design, emergency core cooling, and environmental qualification requirements are not influenced by this modification.

- GDC-50, Containment Design Basis, is described in FSAR Section 3.1.5.1.

  The Containment Building is designed to withstand pressures and temperatures resulting from a spectrum of LOCAs and secondary system ruptures.

  Vital Containment Building subcompartments, such as the steam generator compartment, the reactor cavity, and the pressurizer compartment, are designed to withstand, with a safe margin, peak differential pressures resulting from postulated hot leg, cold leg, and pressurizer line breaks.

  The assumptions used, other details of the containment pressure temperature transient analysis, and the subcompartmental differential pressure analyses are presented in FSAR Section 6.2.1. The structural details are described in FSAR Section 3.8. See FSAR Section 6.2.2 for heat removal; FSAR Section 3.8 for access openings; and FSAR Sections 3.8, 6.2.4, and 8.3 for penetrations.

Additional details that define the CPNPP licensing basis with respect to subcomparmental differential pressure analyses are presented in FSAR Section 6.2.1.2, Containment Subcompartments. All postulated break locations and types are chosen in accordance with Regulatory Guide 1.46 in order to select a design basis rupture yielding the highest mass and energy release rates consistent with break location criteria. As discussed earlier, in accordance with the 1987 revision to GDC-4, the dynamic effects associated with postulated pipe ruptures of primary coolant looping is excluded from the design basis. This exclusion is based on analyses (see LR Section 2.1.6) that apply the leak-before-break methodology to demonstrate the probability of rupturing such piping is extremely low under design basis conditions. In addition, reactor coolant system (RCS) branch line breaks of 10-inch diameter and larger have been eliminated also based on the application of leak-before-break methodology.

2.6.2.2 Technical Evaluation

Introduction

The containment subcompartments were evaluated for their structural response to potential increases in pressure differentials resulting from postulated accidents that are conservatively assumed to initiate at SPU operating conditions. The pressure transient, resulting from postulated accidents, produces a pressure differential, which reaches a maximum value generally within the initial few seconds after blowdown begins.

Description of Analyses and Evaluations

The SPU analyses were performed using the current licensing basis methodology documented in FSAR Section 6.2.1.2, Containment Subcompartments.

Steam Generator Cubicles

For the main steam line break within the steam generator compartment, the current M&E releases utilized for Units 1 and 2 remain unchanged for the SPU since the no load (that is, 0-percent power) case is the limiting condition. Consequently, no further evaluation of the steam line break within the steam generator compartment is required for the SPU.

For the feedwater line break within the steam generator compartment, the current M&E releases utilized for Units 1 and 2 remain unchanged for the SPU since the feedwater operating conditions associated with the SPU are not significantly different from the current Unit 1 and Unit 2 conditions. Consequently, no further evaluation of feedwater line break within the steam generator compartment is required for the SPU.

For the residual heat removal (RHR) line break within the steam generator compartment, the current M&E releases utilized for Unit 1 remain bounding for the SPU. However, since the current M&E releases utilized for Unit 2 increase by approximately 2.5 percent for the SPU, the impact of increase was evaluated to ensure that the Unit 2 steam generator compartments maintain their structural integrity at SPU conditions.

For the auxiliary feedwater line break within the steam generator compartment, the current M&E releases utilized for Unit 1 and Unit 2 remain unchanged for the SPU since the no load (that is, 0-percent power) case is the limiting condition. Consequently, no further evaluation of the auxiliary feedwater line break within the steam generator compartment is required for the SPU.

Main Steam Penetration Area

For the main steam line break within the main steam penetration area, the current M&E releases utilized for Units 1 and 2 remain unchanged for the SPU since the no load (i.e., 0-percent power) case is the limiting condition. Consequently, no further evaluation of the steam line break within the main steam penetration area is required for the SPU.

Feedwater Penetration Area

For the feedwater line break within the feedwater penetration area, the current M&E releases utilized for Unit 1 remains unchanged for the SPU, since the feedwater operating conditions associated with SPU are not significantly different from the current Δ76 steam generator conditions.

However, since the Unit 2 feedwater operating conditions associated with SPU is more limiting than the assumed conditions utilized in the current analyses with the D-5 steam generators, an analysis was performed to ensure that the Unit 2 feedwater penetration area maintains its structural integrity at SPU conditions.

Pressurizer Cubicle

For the spray line break, the SPU loss-of-coolant-accident (LOCA) short-term M&E increase is within the 10-percent margin included in the current evaluations for Unit 2 and documented in FSAR Table 6.2-21. Consequently, no further evaluation of spray line break is required for the SPU for Unit 2.

The original short-term pressurizer spray line break M&E releases for Unit 1 also included the 10-percent margin on the plant-specific release rates. The current analyses supporting steam generator replacement at Unit 1 with the Δ76 model conclude that at least 1 percent of that margin remains available. Since the SPU RCS temperatures and the associated plant-specific instrument uncertainties are slightly less limiting than the conditions utilized for steam generator replacement at Unit 1, the current pressurizer spray line break M&E releases remain bounding with 1-percent margin for the SPU. Consequently, no further evaluation of spray line break within the pressurizer compartment is required for the SPU for Unit 1.

Excess Letdown Heat Exchanger Cubicle

For the chemical and volume control system (CS) line break within the excess letdown heat exchanger cubicle, the current M&E releases utilized for Unit 1 and Unit 2 remain bounding for the SPU since the CS line operating conditions associated with SPU are less limiting. Consequently, no further evaluation of CS line break within the excess letdown heat exchanger cubicle is required for the SPU.

Regenerative Heat Exchanger Cubicle

For the CT line break within the regenerative heat exchanger cubicle, the current M&E releases utilized for Units 1 and 2 remain bounding for the SPU since the CT line operating conditions associated with SPU are less limiting. Consequently, no further evaluation of CT line break within the regenerative heat exchanger cubicle is required for the SPU.

Results

The increase in the Unit 2 RHR line break M&E releases, due to the SPU, results in a conservatively estimated 2.5-percent increase in the pressure differential across the Unit 2 steam generator walls. The design of the steam generator walls has been evaluated for this pressure increase and determined to be bounded by the current design basis. Effects on piping and components within the steam generator compartment are addressed in Licensing Report (LR) subsection 2.2.2.1, NSSS - Pipe and Support (Class 1).

The increase in the Unit 2 feedwater line break M&E releases, due to the SPU, results in an increase of pressure differentials across the Unit 2 feedwater penetration area walls and slabs. For the SPU, a bounding analysis was performed for both units using Unit 1 mass and energy releases. Evaluation of the feedwater penetration area walls and slabs has demonstrated that the current design margin (defined as design strength of structural element greater than load demand on structural element) is sufficient to address the load demand increase as a result of SPU conditions.

2.6.2.3 Conclusion

The evaluation of the subcompartment assessment and the changes in predicted pressurization resulting from the increased mass and energy releases concludes that containment SSCs important to safety will continue to be protected from the dynamic effects resulting from pipe breaks and that the subcompartments will continue to have sufficient margins to prevent fracture of the structure due to pressure difference across the walls following implementation of the proposed SPU. Based on this, Luminant Power concluded that the plant will continue to meet the CPNPP current licensing basis with respect to the requirement of GDCs-4 and -50 for the proposed SPU. Therefore, Luminant Power finds the proposed SPU acceptable with respect to subcompartment analysis.

2.6.3 Mass and Energy Release

2.6.3.1 Mass and Energy Release for Postulated Loss-of-Coolant Accidents

2.6.3.1.1 Regulatory Evaluation

The release of high-energy fluid from pipe breaks could challenge the structural integrity of the containment, including subcompartments and systems within the containment. The review covered the energy sources that are available for release to the containment and the mass and energy (M&E) release rate calculations for the initial blowdown phase of the accident.

The acceptance criteria for M&E release analyses for postulated loss-of-coolant accidents (LOCAs) are based on:

- General Design Criterion (GDC)-50, insofar as it requires that sufficient conservatism is provided in the M&E release analysis to assure that containment design margin is maintained.
- 10 CFR Part 50 Appendix K, insofar as it identifies sources of energy during a loss-of-coolant accident (LOCA).

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the adequacy of assumptions regarding energy sources available for release to the containment and the M&E release rate calculations relative to conformance to GDC-50, Containment Design Basis, is described in FSAR Section 3.1.5.1.

The Containment Building is designed to withstand pressures and temperatures resulting from a spectrum of LOCAs and secondary system ruptures.

Vital Containment Building subcompartments, such as the steam generator compartment, the reactor cavity, and the pressurizer compartment, are designed to withstand, with a safe margin, peak differential pressures resulting from postulated hot leg, cold leg, and pressurizer line breaks.

The assumptions used, the other details of the containment pressure temperature transient analysis, and the subcompartmental differential pressure analysis are presented in FSAR Section 6.2.1. The structural details are described in FSAR Section 3.8. See FSAR Section 6.2.2 for heat removal; FSAR Section 3.8 for access openings; and FSAR Sections 3.8, 6.2.4, and 8.3 for penetrations.

For purposes of evaluating the integrity of the containment as a whole and the integrity of structures internal to the containment (subcompartments), the effects of M&E releases are examined for both long- and short-term releases, respectively.

Primary Containment

Licensing Report (LR) subsection 2.6.1 addresses the primary containment functional design. It discusses the containment LOCA response analysis. In addition, the containment functional design requirements are discussed in FSAR Section 6.2.1.1.1.

FSAR Section 6.2.1.3 provides the current licensing basis analysis regarding M&E releases to the containment subsequent to a LOCA. This analysis identifies the sources of energy available for release to the containment.

Containment Subcompartments

LR subsection 2.6.2 discusses the containment subcompartment analysis in more detail. FSAR Section 6.2.1.2 provides a discussion regarding the short-term M&E release calculations impact on containment subcompartments. In accordance with the 1987 revision to GDC-4 (refer to FSAR Section 3.6B.2), the dynamic effects of RCS main loop piping breaks and reactor coolant system (RCS) branch line breaks 10-inch diameter and larger have been eliminated from consideration.

FSAR Section 6.2.1.2.1 identifies the following limiting breaks for subcompartment analysis:

- Steam Generator Compartment
  - Double-ended rupture (DER) of the main steam line in steam generator compartment #3
  - DER of the main steam line in steam generator compartment #4
  - DER of the main feedwater line in steam generator compartment #4
  - DER of the residual heat removal 6-inch line in steam generator compartment #4
  - DER of the auxiliary feedwater 6-inch line in steam generator compartment #4
- Main Steam Penetration Area
  - DER of the main steam line
- Feedwater Penetration Area
  - DER of the main feedwater line

- Pressurizer Compartment
  - DER of a pressurizer spray line at the nozzle

2.6.3.1.2 Technical Evaluation

2.6.3.1.2.1 Long-Term LOCA M&E Releases

The evaluation/generation of the design basis long-term LOCA M&E release data was completed to support the stretch power uprate (SPU) program operation.

2.6.3.1.2.1.1 Introduction

The long-term LOCA M&E releases are described in the FSAR Section 6.2.1.3. The M&E release rates described in this section form the basis of further computations to evaluate the containment response following the postulated LOCA (FSAR Section 6.2.1.1) and to ensure that containment design margin is maintained.

The uncontrolled release of pressurized high-temperature reactor coolant, termed a LOCA, will result in the release of steam and water into the containment. This, in turn, will result in increases in the local subcompartment pressures and an increase in the global containment pressure and temperature. Therefore, both long-term and short-term effects on the containment resulting from a postulated LOCA were considered using the conditions for CPNPP at the uprated core power.

The long-term LOCA M&E releases analyzed using the Reference 1 methodology for the CPNPP SPU program were analyzed out to one hour. The long-term post reflood releases were calculated by the GOTHIC code and were utilized with the blowdown, reflood and post-reflood transient releases from the Reference 1 methods in the containment integrity analysis (discussed in LR subsection 2.6.1). To demonstrate the acceptability of the containment safeguards systems to mitigate the consequences of a hypothetical large-break LOCA (LBLOCA), the long-term LOCA M&E releases were analyzed to one hour and used as input to the containment integrity analysis with GOTHIC that continued the long-term LOCA mass and energy release to 90 days. The containment safeguards systems must be capable of limiting the peak containment pressure to less than the design pressure, and limiting the temperature excursion to less than the environmental qualification (EQ) acceptance limits.

The SPU analyses were performed using the Westinghouse LOCA M&E Release Model for Containment Design March 1979 Version, described in WCAP-10325 (Reference 1). The NRC review and approval is found in References 1, 3, and 5.

2.6.3.1.2.1.2 Input Parameters, Assumptions, and Acceptance Criteria

Input Parameters and Assumptions

The M&E release analysis is sensitive to the assumed characteristics of various plant systems, in addition to other key modeling assumptions. Where appropriate, bounding inputs are utilized and instrumentation uncertainties are included. For example, the RCS operating temperatures were chosen to bound the highest average coolant temperature range of all operating cases, and a temperature uncertainty allowance was then added (+5.9°F). The RCS pressure in this analysis is based on a nominal value of 2,250 psia, plus an uncertainty allowance (+30 psi). Nominal parameters are used in certain instances. All input parameters are chosen consistent with accepted analysis methodology.

Some of the most critical items are the RCS initial conditions, core decay heat, safety injection flow, and primary and secondary metal mass and steam generator heat release modeling. Specific assumptions concerning each of these items are discussed in the following paragraphs. Tables 2.6.3.1-1 through 2.6.3.1-3 present key data assumed in the analysis.

The core-rated power of 3,633.7 MWt, adjusted for calorimetric error (that is, 100.6 percent of 3,612 MWt), was used in the analysis. (As previously noted, RCS operating temperatures were chosen to bound the highest average coolant temperature range.) The use of higher temperatures is conservative because the initial fluid energy is based on coolant temperatures, which are at the maximum levels attained in steady-state operation. Additionally, an allowance to account for instrument error and dead band was reflected in the initial RCS temperature. As previously discussed, the initial RCS pressure in this analysis was based on a nominal value of 2,250 psia, plus an allowance that accounted for the measurement uncertainty on pressurizer pressure. The selection of 2,280 psia as the limiting pressure is considered to affect blowdown phase results only, since this represents the initial pressure of the RCS. The RCS rapidly depressurizes from this value until the point where it equilibrates with containment pressure.

The rate at which the RCS blows down is initially more severe at the higher RCS pressure. Additionally, the RCS has a higher fluid density at the higher pressure (assuming a constant temperature), and subsequently has a higher RCS mass available for releases. Therefore, 2,250 psia plus uncertainty was selected for the initial pressure as the limiting condition for the long-term M&E release calculations.

The selection of the fuel design features for the long-term M&E release calculation is based on the need to conservatively maximize the energy stored in the fuel at the beginning of the postulated accident (that is, to maximize the core-stored energy). The core-stored energy is based on the time in life for maximum fuel densification. The assumptions used to calculate the fuel temperatures for the core-stored energy calculations account for appropriate uncertainties associated with the models in the PAD code (such as calibration of the thermal model, pellet densification model, or cladding creep model). In addition, the fuel temperatures for the core-stored energy calculation account for appropriate uncertainties associated with manufacturing tolerances (such as pellet as-built density). The total uncertainty for fuel temperature calculation is a statistical combination of these effects and is dependent upon fuel

type, power level, and burnup. Therefore, the analysis very conservatively accounts for the stored energy in the core.

A uniform steam generator tube plugging (SGTP) level of 0 percent was modeled. This assumption maximized the reactor coolant volume and fluid release by including the RCS fluid in all steam generator tubes. During the post-blowdown period, the steam generators are active heat sources since significant energy remains in the secondary metal and secondary mass that has the potential to be transferred to the primary side. The 0-percent tube plugging assumption maximized heat transfer area and, therefore, the transfer of secondary heat across the steam generator tube. Additionally, this assumption reduced the reactor coolant loop resistance, which reduced the ΔP upstream of the break for the pump suction breaks and increased break flow. Therefore, the analysis very conservatively modeled the effects related to SGTP.

The secondary-to-primary heat transfer is maximized by assuming conservative heat transfer coefficients. This conservative energy transfer is ensured by maximizing the initial internal energy of the inventory in the steam generator secondary side. This internal energy is based on full-power operation plus uncertainties.

Following an LBLOCA inside containment, the safety injection system (SIS) operates to reflood the RCS. The first phase of the SIS operation is the passive accumulator injection. Four accumulators are assumed available to inject. When the RCS depressurizes below 728 psia, the accumulators begin to inject. The accumulator injection temperature was conservatively modeled high at 120°F. Relative to the active pumped emergency core cooling system (ECCS) operation, the M&E release calculation considered configurations, component failures, and offsite power assumptions to conservatively bound respective alignments. The cases include a minimum safeguards case (one charging/SI (Chrg/SI) pump, one high-head SI (HHSI) pump, and one low-head SI (LHSI) pump, see Table 2.6.3.1-2), and a maximum safeguards case, two Chrg/SI, two HHSI and two LHSI pumps, see Table 2.6.3.1-3. In addition, a conservative containment backpressure was assumed to bound the GOTHIC calculated results. The assumption of high containment backpressure was shown in Reference 1 to be conservative for the generation of M&E energy releases.

In summary, the following assumptions were employed to ensure that the M&E releases are conservatively calculated, thereby maximizing energy release to containment:

- Maximum expected operating temperature of the RCS (100-percent full-power operation)
- Allowance for RCS temperature uncertainty (+5.9°F)
- Core rated power of 3,612 MWt
- Allowance for calorimetric error (0.6 percent of power)
- Conservative heat transfer coefficients (that is, steam generator primary/secondary heat transfer and RCS metal heat transfer)

- Allowance in core-stored energy for effect of fuel densification
- An allowance for RCS initial pressure uncertainty (+30 psi)
- A total uncertainty for fuel temperature calculation based on a statistical combination of effects and dependent upon fuel type, power level, and burnup
- A maximum containment backpressure from the containment analysis.
- SGTP level (0 percent uniform)
  - Maximizes reactor coolant volume and fluid release
  - Maximizes heat transfer area across the steam generator tubes
  - Reduces reactor coolant loop resistance, which reduces the ΔP upstream of the break for the pump suction breaks and increases break flow

Therefore, based on the previously discussed conditions and assumptions, an analysis of CPNPP was performed for the release of M&E from the RCS in the event of LOCA at 3,612 MWt core power (without uncertainties).

Application of Single-Failure Criterion

An analysis of the effects of the single-failure criterion has been performed on the M&E release rates for each break analyzed. An inherent assumption in the generation of the M&E release is that offsite power is lost with the pipe rupture. This results in the actuation of the emergency diesel generators (EDGs), required to power the SIS. Operating the EDG delays the operation of the SIS that is required to mitigate the transient. This is not an issue for the double-ended hot leg break (DEHL), which is blowdown limited.

Two cases were analyzed to assess the effects of a single failure. The first case assumed minimum safeguards SI flow based on the postulated single failure of an EDG. This assumption results in the loss of one train of safeguards equipment. Therefore, the remaining ECCS was conservatively modeled as: one Chrg/SI pump, one HHSI pump, and one LHSI pump. The other case assumed maximum safeguards SI flow based on no postulated failures that could impact the amount of ECCS flow. The maximum safeguards case was modeled as: two Chrg/SI pumps, two HHSI pumps and two LHSI pumps. The single failure assumption postulated is the failure of one containment spray train. However, this has no impact on the amount of ECCS flow and, therefore, no impact on the M&E release portion of the analysis. The analysis of the cases described provided confidence that the effect of credible single failures is bounded.

Decay Heat Model

American Nuclear Society (ANS) Standard 5.1 was used in the LOCA M&E release model for CPNPP for the determination of decay heat energy. This standard was balloted by the Nuclear Power Plant Standards Committee (NUPPSCO) in October 1978 and subsequently approved. The official standard was issued in August 1979. Table 2.6.3.1-4 lists the decay heat curve used in the CPNPP SPU program M&E release analysis.

Significant assumptions in the generation of the decay heat curve for use in the LOCA M&E release analysis include the following:

- The decay heat sources considered are fission product decay and heavy element decay of U-239 and Np-239.
- The decay heat power from fissioning isotopes other than U-235 is assumed to be identical to that of U-235.
- The fission rate is constant over the operating history of maximum power level.
- The factor accounting for neutron capture in fission products is taken from Table 10 of the ANS Standard 5.1 (Reference 2).
- The fuel is assumed to be at full power for 10^8 seconds.
- The total recoverable energy associated with one fission is assumed to be 200 MeV/fission.
- Two sigma uncertainty (two times the standard deviation) is applied to the fission product decay.

The NRC approved the use of the ANS Standard-5.1, November 1979 decay heat model for the calculation of M&E releases to the containment following a LOCA.

Acceptance Criteria

The long-term cooling criterion was examined. An LBLOCA is classified as an ANS Condition IV event, an infrequent fault. The relevant requirements to satisfy the acceptance criteria are as follows:

- 10CFR50, Appendix A
- 10CFR50, Appendix K, Paragraph I.A

To meet these requirements, the following must be addressed:

- Sources of energy
- Break size and location
- Calculation of each phase of the accident

2.6.3.1.2.1.3 Description of Analyses and Evaluations

Description of Analyses

The evaluation model (EM) used for the long-term LOCA M&E release calculations is the 1979 model described in WCAP-10325 (References 1, 3, and 5). This EM has been reviewed and approved by the NRC. The initial approval letter is included with Reference 1. Further approval is provided in References 3 and 5.

This report section presents the long-term LOCA M&E releases generated in support of the CPNPP SPU program. These M&E releases were used in the containment integrity analysis and qualification temperature evaluation LR subsection 2.6.1, Primary Containment.

The M&E release rates described in this section form the basis of further computations to evaluate the containment following the postulated accident. Discussed in this section are the long-term LOCA M&E releases for the hypothetical double-ended pump suction (DEPS) rupture with minimum safeguards and DEHL rupture cases. The M&E releases and related analysis information for these cases are shown in Tables 2.6.3.1-5 through 2.6.3.1-13. These cases are used for the long-term containment response analyses in LR subsections 2.6.1, Primary Containment and 2.6.5, Containment Heat Removal.

This section presents the analysis for Unit 1. A sensitivity was performed which demonstrated that the Unit 1 analysis bounds Unit 2.

LOCA M&E Release Phases

The containment system receives M&E releases following a postulated rupture in the RCS. These releases continue over a time period, which, for the LOCA M&E analysis, is typically divided into four phases.

- Blowdown – the period of time from accident initiation (when the reactor is at steady-state operation) to the time that the RCS and containment reach an equilibrium state.

- Refill – the period of time when the lower plenum is being filled by the accumulator and ECCS water. At the end of blowdown, a large amount of water remains in the cold legs, downcomer, and lower plenum. To conservatively consider the refill period for the purpose of containment M&E releases, it is assumed that this water is instantaneously transferred to the lower plenum along with sufficient water to completely fill the lower plenum. This allows an uninterrupted release of M&E to containment. Therefore, the refill period is conservatively neglected in the M&E release calculation.

- Reflood – the period of time that begins when water from the lower plenum enters the core and ends when the core is completely quenched.

- Post-Reflood (Reference 1) – the period of time following the reflood phase. At the end of reflood, the core has been recovered with water and the ECCS continues to supply water to the vessel. Depending on the location of the break, the two-phase mixture in the vessel may pass through the steam generator on the broken loop and acquire heat from the stored energy in the secondary system. The methods from Reference 1 are used until 3,600 seconds.

Computer Codes

The WCAP-10325 (Reference 1) M&E release evaluation model comprises M&E release versions of the following codes: SATAN VI, WREFLOOD, FROTH, and EPITOME. These codes were used to calculate the long-term LOCA M&E releases for CPNPP.

SATAN VI calculates the blowdown phase, the first portion of the thermal-hydraulic transient following break initiation, including pressure, enthalpy, density, M&E flow rates, and energy transfer between primary and secondary systems as a function of time.

The WREFLOOD code addresses the portion of the LOCA transient where the core reflooding phase occurs after the primary coolant system has depressurized (blowdown) due to the loss of water through the break and when water supplied by the ECCS refills the reactor vessel and cools the core. The most important feature of WREFLOOD is the steam/water mixing model.

The FROTH code models the post-reflood portion of the transient until the time that the secondary side of the intact loops steam generators has depressurized to the containment design pressure.

EPITOME continues the FROTH post-reflood portion of the transient from the time at which the secondary equilibrates to the containment design pressure to 3,600 seconds. It also compiles a summary of data for the entire transient, including formal instantaneous M&E release tables and M&E balance tables with data at critical times.

Break Size and Location

Generic studies have been performed and documented in Reference 1 with respect to the effect of postulated break size on the LOCA M&E releases. This section presents the M&E releases to the containment subsequent to a hypothetical LOCA. The release rates were calculated to support the SPU program and were calculated for pipe failures at two distinct locations:

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

A third possible location is the cold leg (between the reactor coolant pump and the vessel), but the generic studies that have been documented in Reference 1 have shown that the double-ended break in the cold leg is not limiting for peak calculated containment pressure and temperature.

During the reflood phase, these breaks have the following characteristics. For a cold leg pipe break, all of the fluid that leaves the core must vent through a steam generator and becomes super-heated. However, relative to breaks at other locations, the core flooding rate (and therefore the rate of fluid leaving the core) is low, because all the core vent paths include the resistance of the reactor coolant pump. For a hot leg break, the vent path resistance is relatively low, which results in a high core flooding rate, and the majority of the fluid which exits the core bypasses the steam generators in venting the containment. The pump suction break combines the effects of the relatively high core flooding rate, as in a hot leg break, and steam generator heat addition, as in the cold leg break. As a result, the pump suction breaks yield the highest energy flow rates during the post-blowdown period.

The spectrum of breaks analyzed includes the double-ended hot leg break and double-ended pump suction break with a discharge coefficient of 1.0. Because of the phenomena of reflood as discussed above, the pump suction break location is the worst case for long term containment depressurization. This conclusion is supported by studies presented in Reference 1 which included studies for hot leg and cold leg breaks. The hot leg break is the worst case for containment pressure due to the high short-term blowdown release associated with this break location.

M&E Release Data

Blowdown M&E Release Data

The SATAN VI code was used for computing the blowdown transient.

Table 2.6.3.1-5 presents the calculated M&E release for the blowdown phase of the DEHL break. For the DEHL break M&E release tables, break path 1 refers to the M&E exiting form the reactor vessel side of the break; break path 2 refers to the M&E release exiting from the steam generator side of the break. Table 2.6.3.1-8 presents the calculated M&E releases for the blowdown phase of the DEPS break. The blowdown phase of the DEPS break applies to both the minimum safeguards case and the maximum safeguards case. For the pump suction breaks, break path 1 in the M&E release tables refers to the M&E exiting from the steam generator side of the break. Break path 2 refers to the M&E exiting from the pump side of the break.

Reflood M&E Release Data

The WREFLOOD code is used for computing the reflood transient.

Table 2.6.3.1-9 presents the calculated reflood M&E for the pump suction double-ended rupture, minimum safeguards case.

The transient responses of the principal parameters during reflood are given in Table 2.6.3.1-10.

Post-Reflood M&E Release Data

The long-term M&E releases account for the transfer of the decay heat and the stored energy in the primary and secondary systems to the containment after the end of reflood. The energy for each source term is acquired at the end of reflood from the Westinghouse M&E release analysis. The rate of energy release is determined by a simplified, RCS model that is coupled to the containment volume. Thus, the flow from the vessel to the containment is dependent on the calculated containment pressure. Table 2.6.3.1-11 presents the calculated post-reflood M&E for the pump suction double-ended rupture minimum safeguards case to 3,600 seconds.

Sources of M&E

The sources of mass considered in the LOCA M&E release analysis are given in Table 2.6.3.1-6 and Table 2.6.3.1-12. These sources include the:

- RCS water
- Accumulator water
- Pumped injection (SI)

The energy inventories considered in the LOCA M&E release analysis are given in Table 2.6.3.1-7 and Table 2.6.3.1-13. The energy sources are the following:

- RCS water
- Accumulator water
- Pumped injection (SI)
- Decay heat
- Core-stored energy
- RCS metal (includes steam generator tubes)
- Steam generator metal (includes transition cone, shell, wrapper, and other internals)
- Steam generator secondary energy (includes fluid mass and steam mass)
- Secondary transfer of energy (feedwater into and steam out of the steam generator secondary: feedwater pump coastdown after the signal to close the flow control valve)

The analysis used the following energy reference points:

- Available energy: 212°F; 14.7 psia (energy available that could be released)
- Total energy content: 32°F; 14.7 psia (total internal energy of the RCS)

The M&E inventories are presented at the following times, as appropriate:

- Time zero (initial conditions)
- End-of-blowdown time
- End-of-refill time
- End-of-reflood time
- Time of broken loop steam generator equilibration to pressure setpoint
- Time of intact loop steam generator equilibration to pressure setpoint
- Time of full depressurization (3,600 seconds)

The energy release from the Zirconium-water reaction is considered as part of the WCAP-10325 (Reference 1) methodology. Based on the way that the energy in the fuel is conservatively released to the vessel fluid, the fuel cladding temperature does not increase to the point were the Zirconium-water reaction is significant. For the LOCA M&E calculation, the energy created by the Zirconium-water reaction value is small and is not explicitly provided in the energy balance tables. The energy that is determined is part of the M&E releases, and is therefore already included in the LOCA M&E release.

The sequences of events for the LOCA transients are shown in Tables 2.6.3.1-14 and 2.6.3.1-15.

2.6.3.1.2.1.4 M&E Release Analysis for Postulated LOCA Results

The LOCA M&E releases from accident initiation to the end of reflood, where applicable, have been provided for the DEHL and for the DEPS break cases. Post-reflood M&E releases after 3,600 seconds were calculated internally to the containment model.

The M&E release transients for the limiting transients are presented in Tables 2.6.3.1-5 through 2.6.3-7 for the DEHL case and Tables 2.6.3.1-8 through 2.6.3.1-13 for the DEPS case with minimum ECCS flows.

The results of this analysis (M&E release rate transients) were used in the containment integrity analysis (see LR subsection 2.6.1, Primary Containment).

2.6.3.1.2.1.5 M&E Release Analysis for Postulated LOCA Conclusion

The consideration of the various energy sources listed in LR subsection 2.6.3.1.2.1.2 for the long-term M&E release analysis provides assurance that all available sources of energy have been included in this analysis. By addressing all available sources of energy as well as the limiting break size and location and the specific modeling of each phase of the long-term LOCA transient, the review guidelines presented in SRP, Section 6.2.1.3 have been satisfied.

2.6.3.1.2.2 Short-Term LOCA M&E Releases

An evaluation was conducted to determine the effect of the CPNPP SPU program on the short-term LOCA-related M&E releases that support the subcompartments discussed in FSAR

Section 6.2.1.2. In accordance with the 1987 revision to GDC-4 (Refer to FSAR Section 3.6B.2), the dynamic effects of RCS main loop piping breaks and RCS branch line breaks 10 inch diameter and larger have been eliminated from consideration.

The short-term LOCA-related M&E releases were used as input to the subcompartment analyses (see LR subsection 2.6.2, Subcompartment Analyses). These analyses were performed to ensure that the walls of a subcompartment can maintain their structural integrity during the short pressure pulse (generally less than 3 seconds) accompanying a high-energy line pipe rupture within that subcompartment. Short-term M&E release calculations are performed to support the steam generator compartments, the main steam penetration area, the feedwater penetration area, and the pressurizer compartment (see FSAR Section 6.2.1.2.1).

CPNPP is licensed in accordance with the 1987 revision to GDC-4. This eliminates the need to consider RCS branch lines 10 inches in diameter and greater for subcompartment pressurization. The evaluation showed that the design basis pressurizer spray line break releases would remain bounding. LR subsection 2.6.2, Subcompartment Analyses, discusses the short-term evaluation conducted for this program.

2.6.3.1.2.2.1 Introduction

The containment internal structures are designed for a pressure buildup that could occur following a postulated LOCA. If a LOCA were to occur in these relatively small volumes, the pressure would build up at a faster rate than the overall containment, thus imposing a differential pressure across the walls of the compartments. The evaluation of the containment internal structures is discussed in FSAR Section 6.2.1.2.

Short-term LOCA M&E release calculations are performed to support the steam generator compartments, the main steam penetration area, the feedwater penetration area, and the pressurizer compartment.

CPNPP has been licensed in accordance with the 1987 revision to GDC-4. With the elimination of the large RCS breaks, the only break locations that need to be considered are the largest branch lines off of the primary loop piping. These branch lines include the pressurizer spray line and the residual heat removal (RHR) line from the hot leg to the first isolation valve. The releases associated with these smaller breaks would be considerably lower than the large RCS breaks.

2.6.3.1.2.2.2 Input Parameters, Assumptions, and Acceptance Criteria

Input Parameters and Assumptions

The short-term LOCA M&E release analysis is sensitive to the assumed characteristics of various plant systems, in addition to other key modeling assumptions. Where appropriate, bounding inputs are utilized and instrumentation uncertainties are included. For example, the RCS operating temperatures were chosen to bound the temperature range of all operating cases and a temperature uncertainty allowance (-5.96°F) was then included. The RCS

pressure in this analysis is based on a nominal value of 2,250 psia plus an uncertainty allowance and pressure drop to arrive at a value of 2,308 psia. All input parameters are chosen consistent with accepted analysis methodology. Increased power has no impact on the short-term releases because of the duration of the event (that is, ~3.0 seconds). Only reductions in the initial RCS temperature conditions (including uncertainties) would affect the blowdown M&E release rates and thus the results.

Any possible change in the core-stored energy does not adversely affect the normal plant operating parameters, system actuations, accident mitigating capabilities or assumptions important to the short-term LOCA M&E releases. The SPU program does not create conditions more limiting than those assumed in the current analyses. Any change in core-stored energy would have no effect on the releases because of the short duration of the postulated accident.

Therefore, the only effects that need to be addressed are the change in RCS coolant temperatures and the changes in analysis assumptions for RCS coolant pressure.

In summary, the following assumptions were employed to ensure that the pressurizer spray line break releases were conservatively calculated for the SPU program:

- Minimum RCS vessel/core inlet temperature 542.2°F
- Allowance for RCS temperature uncertainty of -5.96°F
- Allowance for RCS pressure uncertainty of + 30 psi

Acceptance Criteria

The short-term cooling criterion was also examined. A LOCA is classified as an ANS Condition IV event – an infrequent fault. The relevant requirements to satisfy the acceptance criteria are as follows:

The NRC's NUREG-0800, Section 6.2.1.3, M&E Release Analysis for Postulated Loss-of-Coolant Accidents subsection II, Part 3a provides guidance on NRC's expectations for what must be included in a LOCA M&E release calculation, if that calculation is to be acceptable. The Westinghouse M&E models described in WCAP-8264 Rev. 1 (Reference 4) have been found by the NRC to satisfy those expectations.

2.6.3.1.2.2.3 Description of Analysis and Evaluations

Description of Analysis

Short-term releases are linked directly to the critical mass flux, which increases with increasing pressures and decreasing temperatures. The short-term LOCA releases are expected to increase due to changes associated with the current RCS conditions. Short-term blowdown transients are characterized by a peak M&E release rate that occurs during a subcooled condition; thus the Zaloudek correlation, which models this condition, is currently used in the short-term LOCA M&E release analyses (Reference 4). This correlation was used to conservatively evaluate the impact of the deviations in the RCS inlet and outlet temperature for

the SPU program. Therefore, using lower temperatures maximizes the short-term LOCA M&E releases.

As previously stated, CPNPP has been licensed in accordance with the 1987 revision to GDC-4. With the elimination of the large RCS breaks, the only break locations that need to be considered are the largest branch lines off of the primary loop piping. These branch lines include the pressurizer spray line and the RHR line from the hot leg to the first isolation valve. The releases associated with these smaller breaks are considerably lower than the large RCS breaks

Short-term LOCA M&E release calculations are performed to support the steam generator subcompartments, the main steam line penetration area, the feedwater line penetration area, and the pressurizer compartment. A double-ended break in the pressurizer spray line is the limiting break for the pressurizer compartment. Refer to LR subsection 2.6.2 for the analysis of line breaks within the containment subcompartments.

The pressurizer spray line break LOCA M&E analyzed for CPNPP are found in FSAR Table 6.2.1-15 and are found to be the same as Reference 4, Table III-2-5. These mass and energy releases contain 10-percent margin.

The combination of the minimum RCS vessel/core inlet temperature and the temperature uncertainty is such that the current pressurizer spray line break releases remain bounding and valid for the SPU program.

The steam generator compartment RHR line break is addressed in FSAR in Section 6.2.1.2.1, which describes the breaks analyzed for the steam generator compartment. They are as follows:

- Main steam line break (see FSAR Table 6.2.1-100)
- Feedwater line break (see FSAR Table 6.2.1-101)
- RHR 6-inch line (see FSAR Table 6.2.1-98)
- Auxiliary feedwater 6-inch line (see FSAR Table 6.2.1.99)

Refer to LR subsection 2.6.2 for the analysis of the line breaks within the containment subcompartments.

2.6.3.1.2.2.4 Short-Term LOCA M&E Releases Results

In summary, the pressurizer spray line break is not impacted by the RCS conditions for the CPNPP SPU program. The current releases remain bounding and valid. The impact of the SPU program on the compartment response is discussed in LR subsection 2.6.2.

2.6.3.1.2.2.5 Short-Term LOCA M&E Releases Conclusion

The LOCA mass and energy releases presented in the Comanche Peak FSAR Chapter 6.2 have been evaluated to determine the affect of the SPU program on the short term LOCA mass and energy releases. The pressurizer spray line break remains bounding.

2.6.3.1.3 Conclusion

Luminant Power has reviewed the M&E release assessment and concludes that it has adequately addressed the effects of the SPU program and appropriately accounts for the sources of energy identified in 10CFR50, Appendix K. Based on this, Luminant Power finds that the M&E release analysis will meet the CPNPP current licensing basis with respect to the requirements in GDC-50 for ensuring that the analysis is conservative. Therefore, Luminant Power finds the SPU program acceptable with respect to M&E release for postulated LOCA.

2.6.3.1.4 References

1. WCAP-10325-P-A, (Proprietary) and WCAP-10326-A, (Nonproprietary), "Westinghouse LOCA Mass and Energy Release Model for Containment Design, March 1979 Version," May 1983.

2. ANSI/ANS-5.1 1975, "American National Standard for Decay Heat Power in Light Water Reactors," August 1979.

3. Mr. Herbert N. Berkow (NRC) to Mr. J. A. Gresham (W), "Acceptance of Clarifications of Topical Report WCAP-10325, 'Westinghouse LOCA Mass And Energy Release Model For Containment Design – March 1979 Version' (TAC No. MC7980)," October 18, 2005.

4. WCAP-8264, Rev. 1, August 1975 and WCAP-8312, Rev. 2, "Topical Report Westinghouse Mass and Energy Release Data Containment Design."

5. Docket No. 50-315, Amendment No. 126, Facility Operating License No. DPR-58 (TAC No. 71062), for D. C. Cook Nuclear Plant Unit 1, June 9, 1989.

Table 2.6.3.1-1 System Parameters Initial Conditions	
Parameters	**Value**
Nominal Core Thermal Power (MWt)	3,612.0
RCS Total Flow Rate (Lbm/sec)	39,555.6
Vessel Outlet Temperature(1) (°F)	626.3
Core Inlet Temperature(1) (°F)	563.9
Vessel Average Temperature(1) (°F)	595.1
Initial Steam Generator Steam Pressure (psia)	1,032.0
Steam Generator Design: Unit 1 Unit 2	 Δ76 D-5
SGTP (%)	0
Initial Steam Generator Secondary Side Mass (Lbm)	138,371.2
Assumed Maximum Containment Backpressure (psia)	64.7
Accumulator Water volume (ft^3) per accumulator (minimum)(2) N_2 cover gas pressure (psia) (maximum) Temperature (°F)	 818.0 728.0 120.0
SI Start Time, (sec) (total time from beginning of event, which includes the maximum delay from reaching the setpoint)	31.3
Auxiliary Feedwater Flow (gpm/steam generator) (Minimum Safeguards)	0
Auxiliary Feedwater Flow (gpm/steam generator) (Maximum Safeguards)	0

Notes:

RCS total flow rate, RCS coolant temperatures, N2 cover gas pressure, and steam generator secondary side mass include appropriate uncertainty and/or allowance.

1. RCS coolant temperatures include uncertainty of +5.9°F.
2. Does not include accumulator line volume.

Table 2.6.3.1-2 SI Flow Minimum Safeguards	
RCS Pressure (psia)	**Total Flow (gpm)**
Injection Mode (Reflood phase)	
14.7	5,308.9
34.7	4,996.3
54.7	4,683.8
74.7	4,371.1
94.7	4,058.6
114.7	3,746.0
Cold Leg Recirculation Mode	
RCS Pressure (psia)	Total Flow (gpm)
54.7	3,433.5
Hot Leg Recirculation Mode	
RCS Pressure (psia)	Total Flow (gpm)
54.7	3,057.5

Table 2.6.3.1-3 SI Flow Maximum Safeguards	
RCS Pressure (psia)	**Total Flow (gpm)**
Injection Mode (Reflood phase)	
14.7	10,013.2
34.7	9,581.6
54.7	9,122.4
74.7	8,631.6
94.7	8,103.5
114.7	7,530.1
Cold Leg Recirculation Mode	
RCS Pressure (psia)	Total Flow (gpm)
14.7	8,546.0
Hot Leg Recirculation Mode	
RCS Pressure (psia)	Total Flow (gpm)
14.7	5,333.1

Table 2.6.3.1-4 LOCA M&E Release Analysis Core Decay Heat Fraction	
Time (sec)	**Decay Heat Generation Rate (Btu/Btu)**
10	0.053876
15	0.050401
20	0.048018
40	0.042401
60	0.039244
80	0.037065
100	0.035466
150	0.032724
200	0.030936
400	0.027078
600	0.024931
800	0.023389
1,000	0.022156
1,500	0.019921
2,000	0.018315
4,000	0.014781
6,000	0.013040
8,000	0.012000
10,000	0.011262
15,000	0.010097
20,000	0.009350
40,000	0.007778
60,000	0.006958
80,000	0.006424
100,000	0.006021
150,000	0.005323
200,000	0.004847
400,000	0.003770
600,000	0.003201
800,000	0.002834
1,000,000	0.002580
2,000,000	0.001909
4,000,000	0.001355
6,000,000	0.001091
8,000,000	0.000927
10,000,000	0.000808

Table 2.6.3.1-5 DEHL Break Blowdown M&E Release				
	Break Path No.1(1)		Break Path No. 2(2)	
Time Seconds	**Mass lbm/sec**	**Energy Thousand Btu/sec**	**Mass lbm/sec**	**Energy Thousand Btu/sec**
0	0	0	0	0
0.00111	47,522.2	30,918.4	47,520.6	30,916
0.101	43,784.1	28,636.4	28,426	18,454.9
0.201	37,592	24,563	24,848.3	16,031.3
0.301	35,969.4	23,481	22,643.4	14,443.3
0.402	34,793.6	22,716.4	21,467.4	13,511.1
0.502	34,288.1	22,390.3	20,691.9	12,855.9
0.601	34,235.6	22,366.4	20,164.8	12,383
0.701	33,884.2	22,180.3	19,764.7	12,014.1
0.802	33,182.6	21,793.3	19,424.4	11,702.7
0.902	32,491.7	21,428.8	19,154.6	11,452.2
1	31,962.7	21,181.1	18,938.7	11,249.3
1.1	31,733.5	21,141.7	18,721.8	11,057.2
1.2	31,420.1	21,041.9	18,587.1	10,922.3
1.3	30,925	20,812.5	18,510.2	10,828
1.4	30,337.4	20,510.9	18,474.5	10,762.9
1.5	29,766.7	20,205.7	18,471.5	10,719.9
1.6	29,328.2	19,980.6	18,486.9	10,689.7
1.7	28,998.2	19,824.1	18,515.4	10,670
1.8	28,599.2	19,616	18,549.9	10,655.7
1.9	28,067.5	19,305.1	18,580.2	10,642.2
2	27,480.3	18,946.1	18,605.8	10,628.4
2.1	26,974.9	18,639.4	18,627.5	10,615
2.2	26,586.2	18,413.6	18,642.7	10,600.8
2.3	26,199.5	18,185.9	18,647.8	10,583.6
2.4	25,756.5	17,908.8	18,637.8	10,560.6
2.5	25,310.7	17,622.2	18,615.3	10,533
2.6	24,896.4	17,352.7	18,582.8	10,502.3
2.7	24,517	17,104.3	18,540.6	10,468.3
2.8	24,168.3	16,874.1	18,487.7	10,430.6
2.9	23,854.9	16,666.1	18,424.8	10,389.1
3	23,543.5	16,453.5	18,352.3	10,344.1
3.1	23,227.7	16,230.9	18,267.3	10,293.7
3.2	22,952.1	16,033.4	18,172.3	10,239.1
3.3	22,706.5	15,854.9	18,069.2	10,181.3
3.4	22,467.1	15,676.2	17,957.8	10,120.1
3.5	22,244.9	15,506	17,836.2	10,054
3.6	22,055.5	15,357.3	17,704.6	9,983.2
3.7	21,876.2	15,212.8	17,561.9	9,907
3.8	21,702.3	15,068.8	17,403.1	9,822.3
3.9	21,554	14,941.7	17,235	9,732.9

Table 2.6.3.1-5 (cont.) DEHL Break Blowdown M&E Release				
	Break Path No.1(1)		Break Path No. 2(2)	
Time Seconds	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
4	21,422.1	14,825.3	17,061.5	9,641
4.2	21,182.7	14,607.4	16,706.6	9,453.7
4.4	20,994	14,424.1	16,341.1	9,261.2
4.6	20,867.1	14,284.2	15,973.5	9,067.6
4.8	20,811.4	14,203.4	15,611.4	8,876.9
5	20,797.5	14,142.7	15,318.8	8,725.5
5.2	20,833.4	14,094.1	14,988.5	8,550.7
5.4	20,910.3	14,070	14,649.6	8,369.9
5.6	20,994	14,049.1	14,340.2	8,205.4
5.8	21,092.9	14,034.3	14,026.7	8,038.1
6	21,215.7	14,032.8	13,752.5	7,892.9
6.2	21,374.7	14,051.9	13,465.7	7,739.2
6.4	21,567.9	14,087.6	13,171.9	7,580.7
6.6	21,806	14,143.9	12,898.6	7,433.9
6.8	22,127.1	14,248	12,638.3	7,294.1
7	13,407.1	9,629	12,370.7	7,149.7
7.2	16,873.5	11,756.9	12,129.6	7,019.8
7.4	17,037.3	11,815	11,889.6	6,890.4
7.6	17,129.3	11,788.8	11,670.3	6,772.5
7.8	17,233.5	11,761.2	11,451.8	6,654.1
8	17,361.7	11,772.3	11,230	6,533.2
8.2	17,525.8	11,811.4	11,020.8	6,419.4
8.4	17,587.8	11,779.7	10,809.2	6,303.9
8.6	17,721.7	11,800.1	10,602.6	6,191.3
8.8	17,822.8	11,813.6	10,396.3	6,078.9
9	17,982	11,862	10,191.8	5,967.6
9.2	17,752.3	11,662.9	9,988.3	5,857.3
9.4	17,948.8	11,709.8	9,785.1	5,747.3
9.6	18,179.5	11,779.3	9,587	5,640.5
9.8	18,400.3	11,844.1	9,391.3	5,535.5
10	18,691	11,951.1	9,195.2	5,430.3
10.2	19,390.8	12,290.3	8,999.2	5,325.5
10.2	19,403.6	12,297.4	8,997.6	5,324.7
10.4	19,880.1	12,565.5	8,803.3	5,221.1
10.6	20,199.5	12,732	8,608.3	5,117.5
10.8	22,132.1	13,914.2	8,409.5	5,012.2
11	24,690.6	15,492.1	8,204.7	4,904.2
11.2	23,622.2	14,746.6	8,004	4,799.4
11.4	23,074.7	14,339.3	7,781.8	4,682.6
11.6	22,719.2	14,067.5	7,549.1	4,562
11.8	22,489.5	13,905	7,311.6	4,440.8

Table 2.6.3.1-5 (cont.)

DEHL Break Blowdown M&E Release

Time Seconds	Break Path No.1(1) Mass lbm/sec	Break Path No.1(1) Energy Thousand Btu/sec	Break Path No. 2(2) Mass lbm/sec	Break Path No. 2(2) Energy Thousand Btu/sec
12	22,455.7	13,874.7	7,075.4	4,323
12.2	22,231	13,741.9	6,839.4	4,206.8
12.4	21,959.5	13,550.9	6,611.2	4,096.4
12.6	21,720.1	13,384	6,385.8	3,988.4
12.8	21,471.3	13,215.5	6,168.4	3,885.1
13	21,147.2	13,019.5	5,956.7	3,784.2
13.2	20,696	12,761.4	5,738.4	3,679.6
13.4	20,152.1	12,460.7	5,511.8	3,570.8
13.6	9,564.1	6,669.8	5,280.5	3,462.5
13.8	9,793.6	6,714.7	5,020	3,341.7
14	9,955.2	6,759.2	4,773.3	3,235
14.2	10,025.9	6,786.4	4,544.9	3,138.9
14.4	9,991.4	6,777.2	4,330.7	3,041.4
14.6	9,914.9	6,770.4	4,151.6	2,949.8
14.8	9,735.8	6,713.4	4,028.4	2,869.6
15	9,569.4	6,688.1	3,960.7	2,799.7
15.2	9,303.2	6,605.1	3,942.2	2,742.7
15.4	9,043.5	6,530.1	3,962.9	2,704.2
15.6	8,265.4	6,433.6	4,005	2,679.9
15.8	7,326	6,307.3	4,048.2	2,661.3
16	6,664.6	6,155.3	4,085.6	2,646.3
16.2	6,018.3	5,891.4	4,106.3	2,629.8
16.4	5,353.4	5,504.5	4,111.1	2,612.3
16.6	4,709.2	4,954.2	4,092.5	2,589.5
16.8	4,403.8	4,626.2	4,045.9	2,558.7
17	4,205.9	4,388.8	3,971.4	2,521.3
17.2	4,034.7	4,187.8	3,873.6	2,480.8
17.4	3,873.2	4,018.1	3,754.9	2,437.9
17.6	3,749.8	3,861.9	3,618.5	2,393.1
17.8	3,622.8	3,711.9	3,464.6	2,344.8
18	3,467.9	3,549.4	3,300.8	2,295.5
18.2	3,287.3	3,376.4	3,130.6	2,244.4
18.4	3,136.8	3,200.7	2,957.4	2,189.5
18.6	2,978.2	3,045.9	2,779.5	2,129.1
18.8	2,758.9	2,876.3	2,602.1	2,070.7
19	2,573.9	2,728.5	2,426.3	2,018.6
19.2	2,411	2,604.5	2,229.1	1,959.9
19.4	2,262	2,482.6	2,022.8	1,914.4
19.6	2,126.9	2,372.9	1,784.6	1,857.1
19.8	2,000.9	2,262.5	1,537.5	1,776.1
20	1,889.4	2,153.9	1,321.2	1,611.2

Table 2.6.3.1-5 (cont.) DEHL Break Blowdown M&E Release				
	Break Path No.1 (1)		Break Path No. 2 (2)	
Time Seconds	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
20.2	1,801.9	2,053.4	1,142.3	1,408.1
20.4	1,741.8	1,961.2	1,020.6	1,263.8
20.6	1,338.8	1,609.3	952.6	1,183
20.8	1,062.5	1,306.2	836.5	1,039.5
21	914.5	1,136.4	662.4	824.8
21.2	763.8	953	478.9	598
21.4	636.3	795.3	402.8	505.6
21.6	459.9	572.3	281	353.7
21.8	341.6	424.1	270.4	341.2
22	215.4	265	212.4	268
22.2	147.6	181.2	177.6	225.6
22.4	0	0	0	0

Notes:

1. Path 1: M&E exiting from the reactor vessel side of the break.
2. Path 2: M&E exiting from the steam generator side of the break.

Table 2.6.3.1-6 DEHL Break Mass Balance				
Time (Seconds)		**.00**	**22.40**	**22.40+ε[1]**
		Mass (thousand Lbm)		
Initial	In RCS and ACC	781.98	781.98	781.98
Added Mass	Pumped Injection	0	0	0
	Total Added	0	0	0
*****Total Available*****		781.98	781.98	781.98
Distribution	Reactor Coolant	572.12	80.79	98.33
	Accumulator	209.86	152.71	135.17
	Total Contents	781.98	233.5	233.5
Effluent	Break Flow	0	548.46	548.46
	ECCS Spill	0	0	0
	Total Effluent	0	548.46	548.46
*****Total Accountable*****		781.98	781.98	781.96

Note:

1. +ε is used to indicate that the column represents the bottom of core recovery conditions that occurs instantaneously after blowdown.

Table 2.6.3.1-7 DEHL Break Energy Balance				
Time (seconds)		**.00**	**22.40**	**22.40+ε[(1)]**
		Energy (million Btu)		
Initial Energy	In RCS, ACC, S GEN	986.33	986.33	986.33
Added Energy	Pumped Injection	0	0	0
	Decay Heat	0	8.36	8.36
	Heat from Secondary	0	5.46	5.46
	Total Added	0	13.82	13.82
*****Total Available*****		986.33	1,000.15	1,000.15
Distribution	Reactor Coolant	340.8	22.61	23.91
	Accumulator	18.85	13.72	12.42
	Core Stored	23.94	9.35	9.35
	Primary Metal	166.19	156.12	156.12
	Secondary Metal	122.15	118.14	118.14
	Steam Generator	314.39	321.17	321.17
	Total Contents	986.33	641.11	641.11
Effluent	Break Flow	0	358.56	358.56
	ECCS Spill	0	0	0
	Total Effluent	0	358.56	358.56
*****Total Accountable*****		986.33	999.67	999.67

Note:

1. +ε is used to indicate that the column represents the bottom of core recovery conditions that occurs instantaneously after blowdown.

Table 2.6.3.1-8 DEPS Break Blowdown M&E Release				
	Break Path No. 1[1]		Break Path No. 2 Flow[2]	
Time Seconds	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
0	0	0	0	0
0.00109	89,632.3	50,304.5	43,101.2	24,133.7
0.101	42,880	24,068.4	22,028.8	12,321.3
0.202	48,199.5	27,195.4	24,421	13,670.9
0.301	47,722.6	27,093.1	24,528.1	13,742.5
0.402	47,591.5	27,220.5	23,609.3	13,239.3
0.501	47,613.3	27,462.3	22,440.3	12,591.8
0.601	46,213.2	26,893.4	21,448.8	12,039.1
0.701	46,400.9	27,237.4	20,577.9	11,552.6
0.802	46,336.7	27,420.8	19,947.1	11,201
0.902	45,804.7	27,306.9	19,525.8	10,968.6
1	44,835.1	26,913	19,290.9	10,839.3
1.1	43,774.2	26,453.9	19,147.6	10,760.6
1.2	42,801.9	26,033.7	19,072.5	10,719.5
1.3	41,931.2	25,664.8	19,027.6	10,694.8
1.4	41,153.8	25,343.4	18,999.7	10,679.2
1.5	40,429.1	25,044.6	18,987.4	10,672.2
1.6	39,712.4	24,748.3	18,991.6	10,674.6
1.7	38,980.9	24,445.2	19,005.6	10,682.5
1.8	38,215.6	24,128.2	19,007.6	10,683.7
1.9	37,384.5	23,777.7	18,987.2	10,672.2
2	36,449.4	23,372.9	18,963.7	10,659
2.1	35,428	22,919.8	18,932.2	10,641.6
2.2	34,362.5	22,440.5	18,862.9	10,602.8
2.3	33,172.8	21,879.2	18,746.7	10,537.8
2.4	31,999.5	21,316.3	18,589.2	10,449.4
2.5	30,798.8	20,721	18,352.6	10,315.9
2.6	29,628.7	20,132.8	17,928.3	10,077.6
2.7	28,070	19,251.1	17,725.3	9,965
2.8	25,153.3	17,394.1	17,512.9	9,846.5
2.9	23,235.5	16,222.5	17,292.3	9,723.3
3	22,185.5	15,633.8	17,078.9	9,604.4
3.1	20,987.2	14,886.5	16,877	9,492.2
3.2	19,988.7	14,255.3	16,679.5	9,382.5
3.3	19,199.2	13,756.7	16,498.9	9,282.4
3.4	18,473.2	13,288.2	16,329.9	9,188.8
3.5	17,805.8	12,852.1	16,162.2	9,095.9

Table 2.6.3.1-8 (cont.) DEPS Break Blowdown M&E Release				
	Break Path No. 1[1]		Break Path No. 2 Flow[2]	
Time Seconds	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
3.6	17,202.8	12,456.2	16,010.6	9,012.2
3.7	16,680.5	12,112.8	15,874.5	8,937.1
3.8	16,226	11,812.7	15,741.9	8,864
3.9	15,815.7	11,538.8	15,607	8,789.4
4	15,448	11,290.9	15,480.3	8,719.6
4.2	14,867.1	10,894.8	15,253.6	8,594.7
4.4	14,400	10,564.4	15,006.6	8,458.1
4.6	14,037.5	10,297.4	14,794.2	8,340.9
4.8	13,752	10,074.9	14,779	8,338.8
5	13,529.3	9,889.3	16,233.1	9,159.8
5.2	13,346.6	9,726.2	16,012	9,034.5
5.4	13,220.3	9,600.8	15,666.6	8,840.5
5.6	13,158.5	9,518.5	15,620	8,816.2
5.8	13,150.4	9,473.5	15,510	8,755.8
6	13,160.4	9,442.4	15,341.9	8,662.9
6.2	13,184.9	9,424.2	15,191	8,579.6
6.4	13,225.1	9,417.6	15,063.4	8,509.5
6.6	13,308.7	9,438.8	14,983.6	8,466.1
6.8	13,427.9	9,475.9	14,863.6	8,399
7	13,585.4	9,531.3	14,695.4	8,304
7.2	13,781.2	9,607.6	14,550.7	8,222
7.4	14,011.1	9,702.8	14,421.4	8,148.7
7.6	14,273.9	9,817.4	14,278.5	8,067.6
7.8	14,366.9	9,805.9	14,140.1	7,988.8
8	14,209.6	9,760	14,152.9	7,995.7
8.2	13,330.1	9,489.6	13,867.2	7,831.9
8.4	12,315.4	9,048.1	13,714	7,745
8.6	12,049.6	8,903.2	13,593.2	7,676.9
8.8	12,126.7	8,916.3	13,393.2	7,563.2
9	12,199.3	8,910.2	13,214.6	7,461.6
9.2	12,233.3	8,881.5	13,040	7,362.2
9.4	12,300.9	8,878	12,859.4	7,259.4
9.6	12,345.9	8,850.3	12,683	7,158.7
9.8	12,329.1	8,780.7	12,507.8	7,058.9
10	12,294.9	8,706.9	12,346.2	6,966.7
10.2	12,241.2	8,622.6	12,180.3	6,871.9
10.4	12,106.4	8,486.2	12,023.6	6,782.3
10.6	11,880.8	8,299.4	11,889.1	6,705.4

Table 2.6.3.1-8 (cont.) DEPS Break Blowdown M&E Release				
	Break Path No. 1[1]		Break Path No. 2 Flow[2]	
Time Seconds	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
10.8	11,639.6	8,114.8	11,751.9	6,626.6
11	11,401.5	7,938.7	11,618.6	6,550.1
11.2	11,136.8	7,750.9	11,502.3	6,483.3
11.4	10,873	7,573.8	11,387.2	6,417.1
11.6	10,658.5	7,435.8	11,265.8	6,347.1
11.8	10,464.6	7,310.7	11,148.7	6,279.9
12	10,253.1	7,173.2	11,045.3	6,220.4
12.2	10,053.5	7,046.7	10,932.5	6,155.4
12.4	9,874.6	6,933.3	10,822.3	6,092
12.6	9,684	6,810.4	10,722.9	6,034.7
12.8	9,497	6,692.8	10,620.3	5,975.5
13	9,327.7	6,588.4	10,516.9	5,915.7
13.2	9,155.8	6,481.1	10,421.9	5,860.9
13.4	8,982.2	6,373.5	10,327	5,806.2
13.6	8,816.8	6,273.3	10,232.5	5,751.7
13.8	8,667.6	6,184	10,135.5	5,695.9
14	8,521.4	6,095.2	10,044.1	5,643.5
14.2	8,380.1	6,008.7	9,948.1	5,588.7
14.4	8,235.7	5,918.9	9,843.2	5,529.1
14.6	8,079.9	5,821.4	9,723.7	5,461.5
14.8	7,911.6	5,716.4	9,606.7	5,395.9
15	7,740.3	5,607.4	9,486.9	5,329.1
15.2	7,579.2	5,499.2	9,377.5	5,268.3
15.4	7,443.9	5,401.2	9,287.3	5,212.2
15.6	7,332.6	5,314	9,213.9	5,148
15.8	7,236.3	5,235.9	9,200.4	5,101.4
16	7,144.9	5,164.1	9,194.4	5,045.7
16.2	7,050.4	5,095.2	9,244.6	5,014
16.4	6,949.9	5,027.8	9,282.9	4,975.4
16.6	6,842.1	4,961.6	9,338.5	4,951
16.8	6,726.7	4,896.6	9,380	4,927.9
17	6,606.7	4,834.3	9,356.4	4,881.3
17.2	6,482.6	4,774.4	9,325.8	4,838.8
17.4	6,348.9	4,712.8	9,175.3	4,738.7
17.6	6,221.9	4,660.8	9,193.7	4,727.4
17.8	6,099.1	4,614.2	8,877.6	4,553.3
18	5,980.7	4,572.5	9,020.6	4,611.7
18.2	5,877	4,540.1	8,517.1	4,348

Table 2.6.3.1-8 (cont.) DEPS Break Blowdown M&E Release				
	Break Path No. 1[(1)]		Break Path No. 2 Flow[(2)]	
Time Seconds	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
18.4	5,822.8	4,540.2	8,488	4,322
18.6	5,813.9	4,618.1	8,238.7	4,212.5
18.8	5,683	4,712.2	8,215.9	4,180.8
19	5,352.2	4,749.9	7,873.7	3,994.1
19.2	4,939.2	4,736.5	7,742	3,867.1
19.4	4,391.5	4,569.4	7,364.4	3,630.9
19.6	3,892	4,369.1	6,225.8	2,998.5
19.8	3,429.4	4,076.3	11,041	5,196
20	3,038.1	3,722.2	8,589.4	4,118
20.2	2,816.7	3,476.3	4,022.2	1,929.3
20.4	2,563.1	3,175.2	8,252.9	3,570.9
20.6	2,276.1	2,831	7,793.4	3,369.1
20.8	2,088.3	2,606.4	5,198.2	2,256.7
21	1,952.9	2,443	3,637.9	1,579
21.2	1,811.9	2,270.2	2,806.1	1,170.7
21.4	1,662	2,086.1	4,420.9	1,666.6
21.6	1,514.6	1,904.6	6,026.8	2,204.2
21.8	1,384.8	1,744.6	5,350.7	1,932.3
22	1,268.2	1,600	4,511.2	1,612.8
22.2	1,154.5	1,458.5	4,119.9	1,451.8
22.4	1,057.5	1,337.9	3,840.2	1,323.5
22.6	952.7	1,207.1	3,486.8	1,171.6
22.8	857.2	1,087.3	3,061.1	1,001.4
23	778.6	988.6	2,674.1	851.1
23.2	713.1	906	2,300.7	712.5
23.4	667.6	848.8	1,921.6	580
23.6	625.8	796.2	1,549.8	457.1
23.8	576.7	734	1,149.5	332.6
24	525.7	669.4	723.3	206.4
24.2	469.4	598.1	317.3	89.9
24.4	410.9	523.8	26.5	7.5
24.6	351.3	448.1	0	0
24.8	296.6	378.5	0	0
25	240.1	306.7	180.4	51.5
25.2	182.9	233.8	110.3	31.4
25.4	120.5	154.3	74.7	21.2

Table 2.6.3.1-8 (cont.) DEPS Break Blowdown M&E Release				
	Break Path No. 1[1]		Break Path No. 2 Flow[2]	
Time Seconds	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
25.6	70	89.8	53.5	15.2
25.8	17.6	22.7	0	0
26	0	0	0	0

Notes:

1. Path 1: M&E exiting from the steam generator side of the break.
2. Path 2: M&E exiting from the broken loop reactor coolant pump side of the break.

Table 2.6.3.1-9 DEPS Break Reflood M&E Release – Minimum SI				
Time Seconds	Break Path No. 1(1)		Break Path No. 2 Flow(2)	
	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
26	0	0	0	0
26.5	0	0	0	0
26.7	0	0	0	0
26.8	0	0	0	0
26.9	0	0	0	0
27	0	0	0	0
27	12.7	14.9	0	0
27.1	66.4	78.3	0	0
27.2	34.2	40.4	0	0
27.4	34.2	40.3	0	0
27.5	41.7	49.1	0	0
27.6	49.8	58.7	0	0
27.7	55.8	65.8	0	0
27.8	61.3	72.3	0	0
27.9	66.6	78.5	0	0
28	71.5	84.3	0	0
28.1	76.2	89.9	0	0
28.1	78.5	92.6	0	0
28.2	80.7	95.2	0	0
28.3	85.1	100.3	0	0
28.4	89.2	105.2	0	0
28.5	93.3	110	0	0
28.6	97.2	114.6	0	0
28.7	100.9	119	0	0
28.8	104.6	123.4	0	0
28.9	108.2	127.6	0	0
29	111.6	131.7	0	0
29.1	115	135.7	0	0
30.1	145	171.1	0	0
31.1	170.2	200.9	0	0
32.1	584.1	693.5	5,264.2	729.4
32.2	594.8	706.4	5,341.9	755.2
33.1	610.1	724.9	5,441.5	790
34.1	602.5	715.9	5,381.4	784.2
35.1	593.6	705.2	5,308.1	776
35.9	586.1	696.2	5,246.3	768.8
36.1	584.3	694	5,230.6	767

Table 2.6.3.1-9 (cont.) DEPS Break Reflood M&E Release – Minimum SI				
	Break Path No. 1[1]		Break Path No. 2 Flow[2]	
Time Seconds	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
37.1	575	682.8	5,152.4	757.7
38.1	565.9	671.9	5,074.9	748.5
39.1	557	661.3	4,998.9	739.4
40.1	548.4	651	4,924.9	730.5
40.6	544.3	646	4,888.6	726.1
41.1	540.2	641.1	4,852.9	721.8
42.1	532.2	631.6	4,783	713.4
43.1	524.5	622.4	4,715.3	705.2
44.1	517.1	613.5	4,649.6	697.2
45.1	415.3	491.9	3,693	589.7
46.1	409.7	485.2	3,641.6	583.6
46.2	392	464.2	3,399.8	571.9
47.1	561.4	666.5	368	286
48.1	556.4	660.6	365.7	283.2
49.1	538.3	638.9	358	272.9
50.1	520.6	617.7	350.6	262.9
51.1	504.6	598.5	343.9	253.9
51.9	492.3	583.8	338.7	247
52.1	489.3	580.2	337.5	245.4
53.1	474.6	562.6	331.4	237.2
54.1	460.5	545.9	325.6	229.5
55.1	447.1	529.8	320	222.1
56.1	434.2	514.4	314.7	215.1
57.1	421.8	499.6	309.7	208.4
58.1	409.9	485.4	304.8	202.1
59.1	398.5	471.9	300.2	196
60.1	387.6	458.9	295.8	190.2
61.1	377.1	446.4	291.6	184.7
62.1	367.1	434.4	287.6	179.5
63.1	357.4	422.9	283.8	174.4
64.1	348.2	412	280.1	169.7
65.1	339.4	401.4	276.6	165.1
66.1	330.9	391.4	273.3	160.8
66.5	327.6	387.5	272	159.1
67.1	322.8	381.7	270.1	156.7
68.1	315.1	372.5	267.1	152.8
69.1	307.6	363.7	264.2	149.1

Table 2.6.3.1-9 (cont.) DEPS Break Reflood M&E Release – Minimum SI				
	Break Path No. 1(1)		Break Path No. 2 Flow(2)	
Time Seconds	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
70.1	300.6	355.3	261.5	145.5
71.1	293.8	347.3	258.9	142.2
72.1	287.3	339.6	256.4	139
73.1	281.2	332.3	254.1	136
74.1	275.3	325.3	251.8	133.1
75.1	269.7	318.7	249.7	130.4
76.1	264.3	312.3	247.7	127.8
77.1	259.3	306.3	245.8	125.4
78.1	254.4	300.6	244	123.1
79.1	249.8	295.1	242.3	120.9
80.1	245.5	289.9	240.7	118.8
81.1	241.3	285	239.1	116.9
82.1	237.4	280.4	237.7	115
83.1	233.6	275.9	236.3	113.3
84.1	230.1	271.7	235	111.7
85.1	226.8	267.8	233.8	110.1
86.1	223.6	264	232.7	108.7
86.8	221.5	261.5	231.9	107.7
87.1	220.6	260.5	231.6	107.3
89.1	215.1	254	229.6	104.8
91.1	210.2	248.2	227.9	102.6
93.1	205.9	243.1	226.4	100.6
95.1	202.1	238.6	225	98.9
97.1	198.8	234.7	223.8	97.4
99.1	195.9	231.3	222.8	96.1
101.1	193.4	228.3	221.9	95
103.1	191.1	225.6	221.2	94
105.1	189.2	223.4	220.5	93.2
107.1	187.6	221.4	219.9	92.5
109.1	186.2	219.8	219.4	91.8
111.1	185	218.4	219	91.3
112	184.6	217.9	218.8	91.1
113.1	184.1	217.3	218.7	90.9
115.1	183.3	216.3	218.4	90.5
117.1	182.6	215.6	218.1	90.2
119.1	182.1	215	217.9	90
121.1	181.8	214.5	217.8	89.8

Table 2.6.3.1-9 (cont.) DEPS Break Reflood M&E Release – Minimum SI				
	Break Path No. 1(1)		Break Path No. 2 Flow(2)	
Time Seconds	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
123.1	181.5	214.2	217.7	89.6
125.1	181.3	214	217.6	89.5
127.1	181.2	213.9	217.5	89.5
129.1	181.2	213.9	217.5	89.4
131.1	181.2	213.9	217.5	89.4
133.1	181.3	214	217.5	89.4
135.1	181.5	214.2	217.5	89.5
137.1	181.7	214.4	217.6	89.5
139.1	181.9	214.7	217.6	89.6
139.8	182	214.8	217.7	89.6
141.1	182.2	215	217.7	89.7
143.1	182.5	215.4	217.8	89.8
145.1	182.8	215.8	217.9	89.9
147.1	183.2	216.2	218	90
149.1	183.5	216.6	218.1	90.1
151.1	183.9	217	218.2	90.2
153.1	184.2	217.5	218.3	90.4
155.1	184.6	217.9	218.4	90.5
157.1	185	218.4	218.5	90.6
159.1	185.4	218.8	218.6	90.8
161.1	185.8	219.3	218.7	90.9
163.1	186.2	219.8	218.8	91
165.1	186.6	220.2	218.9	91.2
167.1	187	220.7	219	91.3
168.8	187.4	221.1	219.1	91.5
169.1	187.4	221.2	219.1	91.5
171.1	187.8	221.7	219.2	91.6
173.1	188.3	222.2	219.4	91.8
175.1	188.7	222.7	219.5	91.9
177.1	189.1	223.2	219.6	92.1
179.1	189.5	223.7	219.7	92.2
181.1	189.9	224.2	219.8	92.4
183.1	190.4	224.7	220	92.5
185.1	190.8	225.2	220.1	92.7
187.1	191.2	225.7	220.2	92.9
189.1	191.7	226.2	220.3	93
191.1	192.1	226.8	220.5	93.2

Table 2.6.3.1-9 (cont.) DEPS Break Reflood M&E Release – Minimum SI				
	Break Path No. 1[1]		Break Path No. 2 Flow[2]	
Time Seconds	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
193.1	192.5	227.3	220.6	93.3
195.1	193	227.8	220.7	93.5
197.1	193.4	228.3	220.9	93.7
198.9	193.8	228.8	221	93.8

Notes:

1. Path 1: M&E exiting from the steam generator side of the break.
2. Path 2: M&E exiting from the broken loop reactor coolant pump side of the break.

Table 2.6.3.1-10

DEPS - Minimum Safety Injection Principal Parameters During Reflood

Time sec	Temp °F	Flooding Rate in/sec	Carry-over Fraction	Core Height ft	Down-Comer Height ft	Flow Fraction	Total	Injection Accumulator	SI Spill	Enthalpy Btu/Lbm
							(Pounds mass per second)			
26	185.1	0	0	0	0	0.25	0	0	0	0
26.7	182.9	21.236	0	0.53	1.65	0	8,224.8	8,224.8	0	89.82
27	181.2	25.322	0	1.02	1.59	0	8,147.6	8,147.6	0	89.82
28.1	180.1	2.789	0.303	1.5	4.87	0.342	7,843.5	7,843.5	0	89.82
29	180	2.696	0.419	1.62	7.67	0.365	7,655.5	7,655.5	0	89.82
32.2	180.1	5.516	0.635	2.01	16.11	0.633	6,929.4	6,389.6	0	89.68
34.1	180	5.11	0.69	2.29	16.12	0.629	6,580.5	6,044.4	0	89.67
35.9	180.2	4.836	0.713	2.51	16.12	0.627	6,355.7	5,815.5	0	89.67
40.6	181.5	4.409	0.736	3	16.12	0.619	5,871.1	5,320.3	0	89.65
46.2	183.9	3.454	0.743	3.5	16.12	0.548	4,127.7	3,545.7	0	89.56
47.1	184.3	4.415	0.746	3.58	15.83	0.632	546.3	0	0	88
51.9	187.6	3.876	0.747	4	14.25	0.626	561.2	0	0	88
59.1	194.1	3.204	0.746	4.54	12.57	0.615	580.4	0	0	88
66.5	201.8	2.706	0.743	5	11.48	0.601	592.8	0	0	88
76.1	212.2	2.268	0.74	5.51	10.75	0.581	602.3	0	0	88
86.8	222.6	1.973	0.738	6	10.5	0.562	607.8	0	0	88
99.1	232.3	1.796	0.737	6.51	10.63	0.546	610.7	0	0	88
112	240.7	1.713	0.74	7	11.01	0.538	611.9	0	0	88
127.1	249.1	1.679	0.744	7.55	11.58	0.536	612.3	0	0	88
139.1	255	1.675	0.748	7.98	12.06	0.537	612.3	0	0	88
139.8	255.3	1.675	0.748	8	12.09	0.537	612.3	0	0	88
155.1	261.9	1.679	0.754	8.53	12.72	0.539	612.1	0	0	88
168.8	267	1.685	0.759	9	13.27	0.541	611.8	0	0	88
185.1	272.4	1.693	0.765	9.55	13.91	0.544	611.5	0	0	88
198.9	276.4	1.701	0.77	10	14.45	0.547	611.3	0	0	88

Table 2.6.3.1-11 DEPS Break Post-Reflood M&E Release – Minimum SI				
	Break Path No. 1[(1)]		Break Path No. 2 Flow[(2)]	
Time Seconds	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
199	261.6	326.7	359.1	140.4
204	260.2	324.9	360.5	140.5
209	259.9	324.6	360.8	140.4
214	259.6	324.3	361.1	140.3
219	259.3	323.9	361.4	140.1
224	259	323.5	361.7	140
229	258.6	323	362.1	139.9
234	258.2	322.5	362.5	139.8
239	257.8	321.9	363	139.7
244	257.3	321.3	363.4	139.6
249	256.7	320.7	364	139.6
254	257.2	321.3	363.5	139.2
259	256.6	320.5	364.1	139.2
264	255.9	319.7	364.8	139.1
269	255.2	318.8	365.5	139.1
274	255.5	319.1	365.2	138.8
279	254.7	318.1	366	138.8
284	253.8	317	366.9	138.9
289	253.9	317.1	366.8	138.6
294	253.9	317.1	366.8	138.4
299	252.8	315.8	367.9	138.5
304	252.7	315.6	368	138.3
309	252.4	315.3	368.3	138.2
314	252.1	314.8	368.6	138
319	251.6	314.3	369.1	137.9
324	251.1	313.6	369.6	137.9
329	250.4	312.8	370.3	137.8
334	249.7	311.8	371.1	137.8
339	249.6	311.8	371.1	137.6
344	249.5	311.6	371.3	137.4
349	248.3	310.1	372.4	137.5
354	248.6	310.5	372.1	137.2
359	247.9	309.6	372.8	137.2
364	246.9	308.4	373.8	137.2
369	246.6	307.9	374.1	137.1
374	246.7	308.1	374	136.8
379	245.7	306.8	375	136.9
384	245.1	306.1	375.6	136.8

Table 2.6.3.1-11 (cont.) DEPS Break Post-Reflood M&E Release – Minimum SI				
	Break Path No. 1[(1)]		Break Path No. 2 Flow[(2)]	
Time Seconds	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
389	244.7	305.7	376	136.7
394	244.6	305.4	376.2	136.5
399	243.7	304.4	377	136.5
404	243.6	304.2	377.1	136.3
409	243.1	303.6	377.6	136.2
414	242.5	302.9	378.2	136.2
419	241.9	302.1	378.8	136.1
424	241.5	301.7	379.2	136
429	241.2	301.3	379.5	135.9
434	240.7	300.6	380	135.8
439	240.2	300	380.5	135.7
444	239.5	299.1	381.2	135.7
449	95.4	119.2	525.3	174
586.7	95.4	119.2	525.3	174
586.8	99.6	123.5	521.1	173
589	99.5	123.4	521.2	172.9
1,159	86.6	107.2	534.1	166.2
1,159.8	86.6	107.2	385.4	191.7
1,499.8	81.1	100.4	390.9	186.1
1,500	81.1	100.3	390.9	187.4
1,726.8	81.1	100.3	390.9	187.4
1,726.9	77.7	89.4	394.3	93.5
2,000	74.5	85.8	397.5	94
2,000.1	74.5	85.8	397.5	97.3
2,500	71.4	82.2	400.6	97.9
2,500.1	71.4	82.2	400.6	95.5
3,000	68.3	78.6	403.7	96.1
3,000.1	68.3	78.6	403.7	93.7
3,500	65.1	74.9	406.9	94.3
3,500.1	65.1	74.9	406.9	93.4
3,600	64.5	74.2	407.5	93.5

Notes:

1. Path 1: M&E exiting from the steam generator side of the break.
2. Path 2: M&E exiting from the broken loop reactor coolant pump side of the break.

	Time (Seconds)	.00	26.00	26.00+ε	198.91	586.8	1,726.77	3,600
		Mass (Thousand LBM)						
Initial	In RCS and Accumulator	781.98	781.98	781.98	781.98	781.98	781.98	781.98
Added Mass	Pumped Injection	0	0	0	100.57	341.28	964.56	1,848.72
	Total Added	0	0	0	100.57	341.28	964.56	1,848.72
*** **Total Available** ***		781.98	781.98	781.98	882.55	1,123.26	1,746.54	2,630.7
Distribution	Reactor Coolant	572.12	58.27	81.71	140.89	140.89	140.89	140.89
	Accumulator	209.86	145.22	121.79	0	0	0	0
	Total Contents	781.98	203.5	203.5	140.89	140.89	140.89	140.89
Effluent	Break Flow	0	578.47	578.47	730.16	970.87	1,594.09	2,478.26
	ECCS Spill	0	0	0	0	0	0	0
	Total Effluent	0	578.47	578.47	730.16	970.87	1,594.09	2,478.26
*** **Total Accountable** ***		781.98	781.96	781.96	871.05	1,111.77	1,734.98	2,619.15

Table 2.6.3.1-12

DEPS Break Energy Balance Minimum Safeguards

Note: +ε is used to indicate that the column represents the bottom of core recovery conditions which occurs instantaneously after blowdown.

Table 2.6.3.1-13 **DEPS Break Energy Balance Minimum Safeguards**								
	Time (Seconds)	**.00**	**26.00**	**26.00+ε**	**198.91**	**586.8**	**1,726.77**	**3,600**
		Energy (Thousand Btu)						
Initial Energy	In RCS, Accumulators and Steam Generators	986.33	986.33	986.33	986.33	986.33	986.33	986.33
Added Mass Energy	Pumped Injection	0	0	0	8.85	30.03	107.15	261.29
	Decay Heat	0	8.64	8.64	29.96	66.92	152.03	261.97
	Heat from Secondary	0	5.15	5.15	5.15	5.15	5.15	5.15
	Total Added	0	13.79	13.79	43.97	102.11	264.33	528.41
*** **Total Available** ***		986.33	1,000.11	1,000.11	1,030.29	1,088.43	1,250.65	1,514.73
Distribution	Reactor Coolant	340.8	12.87	14.97	37.35	37.35	37.35	37.35
	Accumulator	18.85	13.04	10.94	0	0	0	0
	Core Stored	23.94	11.84	11.84	4.91	4.71	4.19	3.33
	Primary Metal	166.19	157.43	157.43	131.03	96.65	66.5	51.07
	Secondary Metal	122.15	119.21	119.21	109.74	89.36	56.18	42.5
	Steam Generator	314.39	324.63	324.63	295.5	235.9	142.81	107.11
	Total Contents	986.33	639.01	639.01	578.53	463.97	307.02	241.37
Effluent	Break Flow	0	360.53	360.53	440.45	613.16	921.25	1,251.86
	ECCS Spill	0	0	0	0	0	0	0
	Total Effluent	0	360.53	360.53	440.45	613.16	921.25	1,251.86
*** **Total Accountable** ***		986.33	999.54	999.54	1,018.98	1,077.12	1,228.27	1,493.23
Note: +ε is used to indicate that the column represents the bottom of core recovery conditions which occurs instantaneously after blowdown.								

Table 2.6.3.1-14 Double-Ended Hot Leg Break Sequence of Events	
Time (sec)	**Event Description**
0.0	Break Occurs and Loss of Offsite Power Are Assumed
2.2	Compensated Pressurizer Pressure Reactor Trip for Turbine Trip – 1,845 psig Reached
3.9	Low-Pressurizer Pressure SI Setpoint – 1,700.3 psig Reached – Feedwater Isolation Signal
10.91	Feedwater Isolation Valves Closed
12.4	Broken Loop Accumulator Begins Injecting Water
12.4	Intact Loop Accumulator Begins Injecting Water
22.4	End-of-Blowdown Phase
22.4	Accumulator Mass Adjustment for Refill Period
22.4	Transient Modeling Terminated

Table 2.6.3.1-15 Double-Ended Pump Suction Break - Minimum Safeguards Sequence of Events	
Time (sec)	**Event Description**
0.0	Break Occurs and Loss-of-Offsite Power Are Assumed
2.6	Compensated Pressurizer Pressure Reactor Trip for Turbine Trip (1,845 psig) Reached
4.3	Low Pressurizer Pressure SI Setpoint (1,700.3 psig) Reached – Feedwater Isolation Signal
11.31	Feedwater Isolation Valves Closed
15.1	Broken Loop Accumulator Begins Injecting Water
15.3	Intact Loop Accumulator Begins Injecting Water
26.0	End of Blowdown Phase
26.0	Accumulator Mass Adjustment for Refill Period
31.30	Pumped Safety Injection Begins (Includes 27 Second Diesel Delay)
44.76	Broken Loop Accumulator Water Injection Ends
46.11	Intact Loop Accumulator Water Injection Ends
198.9	End-of-Reflood Phase
449.0	M&E Release Assumption: Broken Loop Steam Generator (SG) Equilibration When the Secondary Temperature is the Saturation (Tsat) At Containment Design Pressure of 50 psig
586.8	M&E Release Assumption: Broken Loop SG Equilibration at Containment Pressure of 40 psig
1,159.8	Switchover to Cold Leg Recirculation Begins
1,572.4	M&E Release Assumption: Intact Loop SG Equilibration When the Secondary Temperature is the Saturation (Tsat) at Containment Design Pressure of 50 psig
1,726.8	M&E Release Assumption: Intact Loop SG Equilibration at Containment of 30 psig
3,600.0	End of Transient

2.6.3.2 Mass and Energy Release for Postulated Secondary-System Pipe Ruptures

2.6.3.2.1 Regulatory Evaluation

Luminant Power review covered the energy sources that are available for release to the containment, the mass and energy release rate calculations, and the single failure analysis performed for steam and feedwater line isolation provisions, which would limit the flow of steam or feedwater to the assumed pipe rupture. The acceptance criteria for mass and energy release analyses for postulated secondary system pipe ruptures are based on:

- General Design Criterion (GDC)-50, insofar as it requires that the margin in the design of the containment structure reflect consideration of the effect of potential energy sources that have not been included in the determination of peak conditions, the experience and experimental data available for defining accident phenomena and containment response, and the conservatism of the model and the value of input parameters.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the adequacy of assumptions regarding energy sources available for release to the containment and the mass and energy release rate calculations relative to conformance to GDC-50, Containment Design Basis, is described in FSAR Section 3.1.5.1. The assumptions used and the details of containment pressure temperature transient analysis for a postulated secondary system pipe rupture is presented in FSAR Section 6.2.1.4.

2.6.3.2.2 Technical Evaluation

2.6.3.2.2.1 Introduction

Steam line ruptures occurring inside the reactor containment structure may result in significant releases of high-energy fluid to the containment environment, producing elevated containment temperatures and pressures. The magnitude of the releases following a steam line rupture is dependent upon the plant initial operating conditions and the size of the rupture as well as the configuration of the plant steam system and the containment design. The CPNPP steam line break and containment response analysis considers a spectrum of cases that vary the initial plant condition, break size, and postulated single failure.

The analysis inputs, assumptions, methods, and acceptance criteria (along with a description of the analyses and evaluation pertaining to the steam line break mass and energy releases inside containment) are presented in the following subsections. The analysis presented is for Unit 1. An evaluation demonstrated that the Unit 1 peak containment analysis results are greater than those calculated for Unit 2, and therefore, the Unit 1 analysis bounds the Unit 2 analysis.

2.6.3.2.2.2 Input Parameters, Assumptions, and Acceptance Criteria

The 16 cases included in the analysis for the SPU have been chosen based on the selection of similar steam line ruptures included in the analyses presented in FSAR Section 6.2.1.4. The cases, listed in Table 2.6.3.2-1 and described in Licensing Report (LR) subsection 2.6.3.2.2.2.5, have been analyzed at the uprated nuclear steam supply system (NSSS) power. Other assumptions regarding important plant conditions and features for the secondary-side assumptions, reactor coolant system (RCS) assumptions, and protection systems actuations are discussed in the following subsections.

2.6.3.2.2.2.1 Secondary-Side Assumptions

This subsection summarizes the major input assumptions associated with the main feedwater system, the auxiliary feedwater system, the steam generator, and the main steam system.

Main Feedwater System

The rapid depressurization that occurs following a steam line rupture typically results in large amounts of water being added to the steam generators through the main feedwater system. A rapid-closing main feedwater isolation valve (FIV) and feedwater control valve (FCV) in each of the main feedwater lines limits this effect. The feedwater addition to the faulted steam generator is maximized to be conservative because it increases the water mass inventory that will be converted to steam and released from the break.

Following the initiation of the steam line break, assumptions are made to ensure that main feedwater flow is conservatively modeled assuming that sufficient feedwater flow is provided to match or exceed the steam flow prior to reactor trip. The initial increase in feedwater flow is in response to the FCV opening in response to the steam flow/feedwater flow mismatch or the decreasing steam generator water level, as well as due to a lower backpressure on the feedwater pump as a result of the depressurizing steam generator. This maximizes the total mass addition prior to feedwater isolation. The feedwater isolation response time, following the safety injection signal, is assumed to be a total of 7 seconds, accounting for delays associated with signal processing plus FIV stroke time. For the circumstance in which the FIV in the faulted loop fails to close, there is no effect on the feedwater isolation response time since the total delay for the FCV closure is also 7 seconds.

Following feedwater isolation, as the steam generator pressure decreases, some of the fluid in the feedwater lines downstream of the isolation or control valve may flash to steam if the feedwater temperature exceeds the saturation temperature. This unisolable feedwater line volume (547 ft^3) is an additional source of fluid that can increase the mass discharged out of the break. The unisolable volume in the feedwater lines is maximized for the faulted loop.

Auxiliary Feedwater System

Generally, within the first minute following a steam line break, the auxiliary feedwater (AFW) system is initiated on any one of several protection system signals. Addition of AFW to the

steam generators will increase the secondary mass available for release to containment as well as increase the heat transferred to the secondary fluid. The AFW flow to the faulted and intact steam generators has been assumed to be a constant flow independent of the faulted steam generator pressure. A maximum flow to the faulted loop of 1,380 gpm is assumed, and operator action is credited to terminate the AFW flow to the faulted steam generator after 10 minutes.

The volume of the AFW piping is minimized, and purging of the AFW piping is not assumed because a minimum volume permits the colder AFW (120°F) to be injected into the steam generator rather than any hotter AFW resident in the piping. The injected AFW is more dense at the lower temperature, which results in a greater mass addition to the faulted steam generator.

Initial Steam Generator Fluid Inventory

A maximum initial steam generator mass in the faulted-loop steam generator has been used in all of the analyzed cases. The use of a high faulted-loop initial steam generator mass maximizes the steam generator inventory available for release to containment. The initial level corresponds to 77-percent narrow-range span (NRS) at all power levels. This consists of a nominal level of 67-percent NRS plus a steam generator water level control uncertainty of 10-percent NRS.

Break Flow Model and Quality of the Break Effluent

Piping discharge resistances are not included in the calculation of the releases resulting from the steam line ruptures (Moody Curve for an f (ℓ/D) = 0 is used).

The quality of the break effluent is assumed to be 1.0, corresponding to saturated steam that is all vapor with no liquid. Although it is expected that there would be a significant quantity of liquid in the break effluent for a full double-ended rupture, the all-vapor assumption conservatively maximizes the energy addition to the containment atmosphere.

Unisolable Steam Line Volume

The steam in the steam line between the break and the main steam line isolation valve (MSIV) is included in the mass and energy released from the break. For most cases, this is the MSIV on the faulted loop. For cases that postulate the faulted-loop MSIV to fail open (see LR subsection 2.6.3.2.2.2.5.3), the model considers the steam in the steam line to each of the MSIV on the intact loops.

2.6.3.2.2.2.2 Reactor Coolant System Assumptions

While the mass and energy released from the break is determined from the assumptions previously mentioned, the rate at which the release occurs is largely controlled by the conditions in the RCS. The major features of the primary-side analysis model are summarized.

- Upon steam line isolation, the intact steam generators may become sources of energy that can be transferred to the faulted steam generator. This energy transfer occurs via

the primary coolant. The reverse heat transfer from the intact steam generators to the RCS is modeled as the temperature in the RCS falls below the intact steam generator fluid temperature.

- Minimum safety injection flow rates corresponding to the failure of one safety injection train have been assumed in this analysis. (This is in addition to the steam line break related single failures noted in LR subsection 2.6.3.2.2.2.5.3.) The flow rates are modeled to conservatively minimize the amount of boron that provides negative reactivity feedback. The delay time to achieve full safety injection flow is assumed to be 27 seconds for this analysis.

- Continued operation of the reactor coolant pumps is assumed to maintain a high heat transfer rate to the steam generators.

- The model includes consideration of the heat that is stored in the RCS metal.

- Core residual heat generation is assumed based upon the 1979 American Nuclear Society ANS Decay Heat + 2σ model (Reference 1).

- Conservative core reactivity coefficients corresponding to end-of-cycle conditions with the most reactive rod stuck out of the core are assumed. This maximizes the reactivity feedback effects as the RCS cools down as a result of the steam line break.

- All cases have credited a minimum shutdown margin of 1.3-percent Δk.

- RCS average temperature is the full-power nominal value of 589.2°F plus an uncertainty of 6.0°F.

- The assumed full-power NSSS power is 3,628 MWt.

2.6.3.2.2.2.3 Protection System Actuations

The protection systems available to mitigate the effects of a steam line break inside containment include reactor trip, safety injection, steam line isolation, and main feedwater isolation. The protection system actuation signals, associated setpoints, and delays that have been modeled in the analysis are provided. The setpoints used are conservative values with respect to the plant-specific values delineated in the Technical Specifications.

The first signal is generated by a safety injection low steam line pressure signal for all double-ended rupture (DER) cases (see LR subsection 2.6.3.2.2.2.5.2). The assumed setpoint is 395.0 psia, with a lead/lag of 10/5 seconds. For split rupture cases (also see LR subsection 2.6.3.2.2.2.5.2), the first signal that is credited comes from the safety injection Hi-1 containment pressure setpoint (5.0 psig). The safety injection signal will do the following:

- Start the safety injection pumps
- Trip the reactor

- Start the AFW pumps
- Close the feedwater isolation valves
- Trip the main feedwater pumps

Either a low steam line pressure safety injection signal or a Hi-2 containment pressure signal (8.0 psig) will result in steam line isolation. Each steam line has a quick-acting isolation valve which is designed to stop flow to prevent uncontrolled steam release from more than one steam generator. A delay time of 7 seconds that accounts for the delays associated with the signal processing plus MSIV stroke time has been assumed.

2.6.3.2.2.2.4 Acceptance Criteria

The main steam line break is classified as an ANS Condition IV event, an infrequent fault. The acceptance criteria associated with the steam line break event resulting in mass and energy releases inside containment is based on an analysis that provides sufficient conservatism to show that the containment design margin is maintained. The specific criteria applicable to this analysis are related to the assumptions regarding power level, stored energy, the break flow model, main and auxiliary feedwater flow, steam line and feedwater isolation, and single failure such that the containment peak pressure and temperature are maximized. These analysis assumptions have been included in this steam line break mass and energy release analysis as discussed in Reference 2.

The specific acceptance criteria for the containment response are discussed in LR subsection 2.6.1.

2.6.3.2.2.2.5 Description of Analyses and Evaluations

The Westinghouse steam line break mass and energy release methodology was approved by the Nuclear Regulatory Commission (NRC) (Reference 3) and is documented in WCAP-8822 (Reference 2). WCAP-8822 forms the basis for the assumptions used in the calculation of the mass and energy releases resulting from a steam line rupture. WCAP-8822 used MARVEL as the mass and energy release system code. This was subsequently replaced by LOFTRAN (References 2 and 4), which was used in the previous CPNPP licensing-basis analysis. However, the analysis documented herein uses the RETRAN code that is documented in WCAP-14882 (Reference 5).

The following limitations in the NRC Safety Evaluation Report (SER) for WCAP-14882 have been adhered to in the use of RETRAN to analyze this event:

- The break flow is the Moody model.

- Only steam (dry vapor) will exit the break, since perfect steam separation in the steam generators is assumed.

- Any superheated conditions will be reset to be equal to the saturation temperature.

As previously mentioned, there are many factors that influence the quantity of and rate of the mass and energy release from the steam line. To encompass these factors, a spectrum of cases is defined that varies the initial power, break definition, and the single failure (see Table 2.6.3.2-1).

2.6.3.2.2.2.5.1 Initial Power Level

The power level at which the plant is operating when the steam line break is postulated can cause different competing effects. A single power level cannot be specified as the most limiting. At higher power levels, there is less initial water/steam in the steam generator, which is a benefit. However, at higher power levels there is a higher initial feedwater flow rate, higher feedwater temperature, higher decay heat, and there is a higher rate of heat transfer from the primary side, which are all penalties. Therefore, representative power levels of 100.6, 70, 30, and 0 percent of the uprated full NSSS power conditions have been investigated for CPNPP based on the information in WCAP-8822 (Reference 2). A calorimetric uncertainty of 0.6 percent is applied to the initial condition for the full power case.

2.6.3.2.2.2.5.2 Break Definitions

At plant power levels of 100.6, 70, 30, and 0 percent of nominal full-load NSSS power, two limiting break sizes have been defined. These break areas are defined as the following:

- A full DER immediately downstream of the flow restrictor in one steam line. Note that a DER is defined as a rupture in which the steam pipe is completely severed and the ends of the break fully displace from each other. The full DER represents the largest break of the main steam line producing the highest mass flow rate from the faulted-loop steam generator and conservatively bounds the plant response to a smaller break size. The effective forward break area is limited by the 1.4 ft^2 cross-sectional area of the flow restrictor that is integral to the steam generator. The actual break area is the cross-sectional area of the pipe, which is 4.75 ft^2.

- A split rupture, the largest break that will neither generate a steam line isolation signal from the primary protection equipment nor result in water entrainment in the break effluent as discussed in Reference 2. Reactor protection and safety injection actuation functions are obtained from containment pressure signals.

2.6.3.2.2.2.5.3 Single-Failure Assumptions

In a manner consistent with the standard approach for licensing-basis analyses, various single failures have been identified and used in the analyzed steam line break mass and energy release cases. The single failures either reduce the heat removal capacity of the containment safeguards system or increase the energy released from the steam line break. The analysis cases separately consider the single failure at each initial power level. The postulated single failures (also discussed in Reference 2) are discussed.

- Containment Safeguards Failure – This is a failure of one safeguards train. The main impact is on the containment response analysis, where the active heat removal is reduced by the loss of one train of containment sprays. All cases will also assume the loss of one train of safety injection flow in the steam line break mass and energy release analysis, which is a conservative assumption. Note that analyses have sometimes referred to this as a diesel failure. A diesel failure will also eliminate the same number of containment spray pumps as a containment safeguards failure. However, a diesel failure has longer delays (from setpoints being reached to actuation of the heat removal systems) and is only relevant when there is a loss of offsite power. A loss of power is a large benefit to the mass and energy releases because the trip of the reactor coolant pumps significantly decreases the primary-to-secondary side heat transfer rate. The benefit of the reactor coolant pumps tripping is much larger than the penalty of the diesel-starting delay for the safeguards systems. Therefore, the bounding steam line break/containment response scenarios are with offsite power available, and diesels do not factor into the plant response.

- Failure of the MSIV in the Faulted Loop – The main steam line isolation function is accomplished via the MSIV in each of the four steam lines. Each valve closes on an isolation signal to terminate steam flow from the associated steam generator. The main steam line rupture upstream of this valve, as postulated for the inside containment analysis, creates a situation in which the steam generator on the faulted loop cannot be isolated, even when the MSIV successfully closes. The break location allows a continued blowdown from the faulted-loop steam generator until it is empty and all sources of main feedwater and auxiliary feedwater addition are terminated. If the faulted-loop MSIV fails to close, the steam header is no longer isolated from the break, and the header's inventory is then blown down via reverse flow out of the break. Isolation of the break occurs due to the closure of the MSIVs on the intact loops.

- Failure of the FIV in the Faulted Loop – The FIVs are the fast-acting (7-second closure time), primary method credited for terminating feedwater addition to the faulted steam generator during a steam line break. If the FIV in the feedline to the faulted steam generator is assumed to fail open, backup isolation is provided via the FCV closure (also a 7-second closure time). The inventory between the FIV and the FCV in the faulted loop, plus any additional pumped main feedwater until FCV closure, would be available to be released to containment. For CPNPP the piping volume between the FIV and the FCV is small; and the valve closure time of each valve is identical. Therefore, the mass and energy releases inside containment for the SPU program conservatively

assume the failure of the FIV in the same loop as the ruptured steam line for all steam line break cases analyzed.

- Failure of the AFW Runout Control Function – The failure of the AFW runout control function results in an increased AFW flow rate to the faulted steam generator. For the CPNPP units, flow to the faulted loop from AFW is limited by passive flow restrictors. Failure of these passive flow restrictors is not considered to be credible. Therefore, this single failure is not applicable to the steam line break mass and energy release inside containment analysis for CPNPP.

2.6.3.2.2.2.6 Results

Using the steam line break analysis methodology documented in Reference 2 as a basis, including CPNPP plant-specific parameter changes associated with the SPU, the mass and energy release rates for each of the sixteen steam line break cases defined in Table 2.6.3.2-1 have been generated for use in containment pressure and temperature response analyses.

The limiting containment pressure case is a 4.7 ft^2 split break initiated from 30-percent power assuming a containment safeguards failure. The break flow rate is shown in Figure 2.6.3.2-1, and the break enthalpy is in Figure 2.6.3.2-2. See LR subsection 2.6.1 for the basis of this case being the limiting transient. LR subsection 2.6.1 also contains a sequence of events for this case, including primary, secondary, and containment system actuations.

A sensitivity was done to consider the effects of the plant response for Unit 2 versus Unit 1. It was found that the plant response to this event is similar for either unit.

2.6.3.2.3 Conclusions

Luminant Power has reviewed the mass and energy release assessment for the postulated secondary system pipe ruptures and finds that the analyses adequately address the effects of the proposed uprate. Luminant Power concludes that the analysis meets the CPNPP current licensing-basis requirements with respect to General Design Criterion (GDC)-50 for ensuring that the analysis is conservative (that is, the analysis includes sufficient margin). Therefore, Luminant Power finds the proposed SPU is acceptable with respect to mass and energy releases for postulated secondary system pipe rupture.

2.6.3.2.4 References

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

2. WCAP-8822 and WCAP-8860, "Mass and Energy Releases Following a Steam Line Rupture," September 1976; WCAP-8822 and WCAP-8860, "Supplement 1 – Calculations of Steam Superheat in Mass/Energy Releases Following a Steam Line Rupture," September 1986; WCAP-8822 and WCAP-8860, "Supplement 2 – Impact of

Steam Superheat in Mass/Energy Releases Following a Steam Line Rupture for Dry and Subatmospheric Containment Designs," September 1986.

3. Letter from Cecil O. Thomas (NRC), "Acceptance for Referencing of Licensing Topical Report WCAP-8821 and WCAP-8859, 'TRANFLO Steam Generator Code Description,' and WCAP-8822 and WCAP-8860, 'Mass and Energy Release Following a Steam Line Rupture,'" August 1983.

4. WCAP-7907, "LOFTRAN Code Description," April 1984.

5. WCAP-14882, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," April 1999.

Table 2.6.3.2-1 Case Definitions Steam Line Break Mass and Energy Releases Inside Containment			
Power Level	**Break Type**	**Break Size (ft^2)**	**Single Failure**
100.6%	DER	1.4	Containment safeguards and FIV
70%	DER	1.4	Containment safeguards and FIV
30%	DER	1.4	Containment safeguards and FIV
0%	DER	1.4	Containment safeguards and FIV
100.6%	Split	4.3	Containment safeguards and FIV
70%	Split	4.5	Containment safeguards and FIV
30%	Split	4.7	Containment safeguards and FIV
0%	Split	4.7	Containment safeguards and FIV
100.6%	DER	1.4	MSIV and FIV
70%	DER	1.4	MSIV and FIV
30%	DER	1.4	MSIV and FIV
0%	DER	1.4	MSIV and FIV
100.6%	Split	4.3	MSIV and FIV
70%	Split	4.5	MSIV and FIV
30%	Split	4.7	MSIV and FIV
0%	Split	4.7	MSIV and FIV

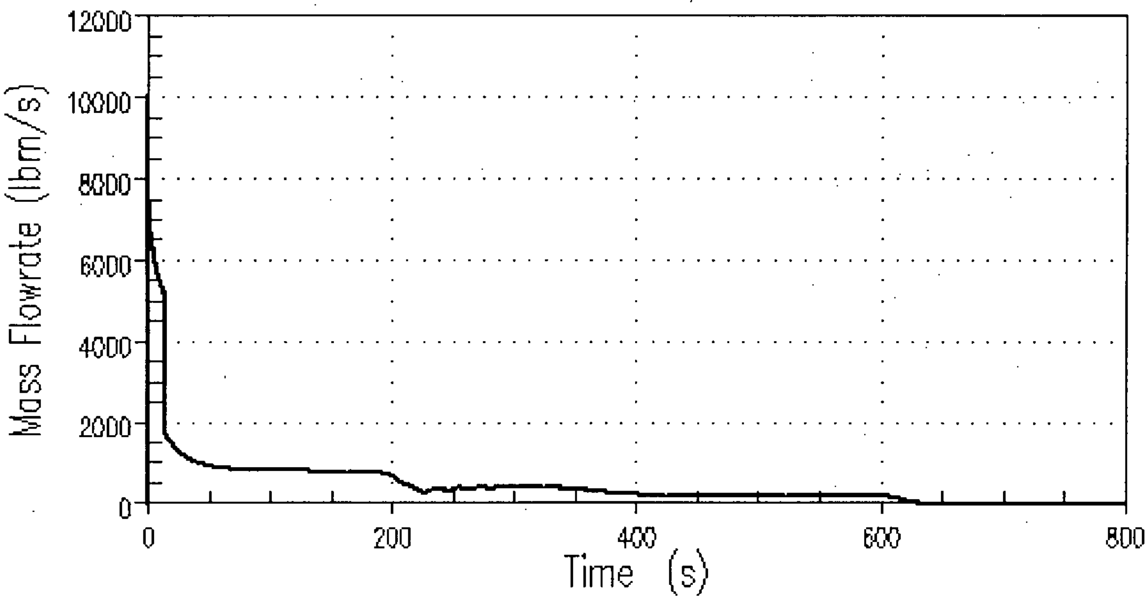

Figure 2.6.3.2-1 **Steam Line Break Mass Release to Containment 4.7 ft^2 Split Break, 30% Power, Containment Safeguards, and FIV Failure**

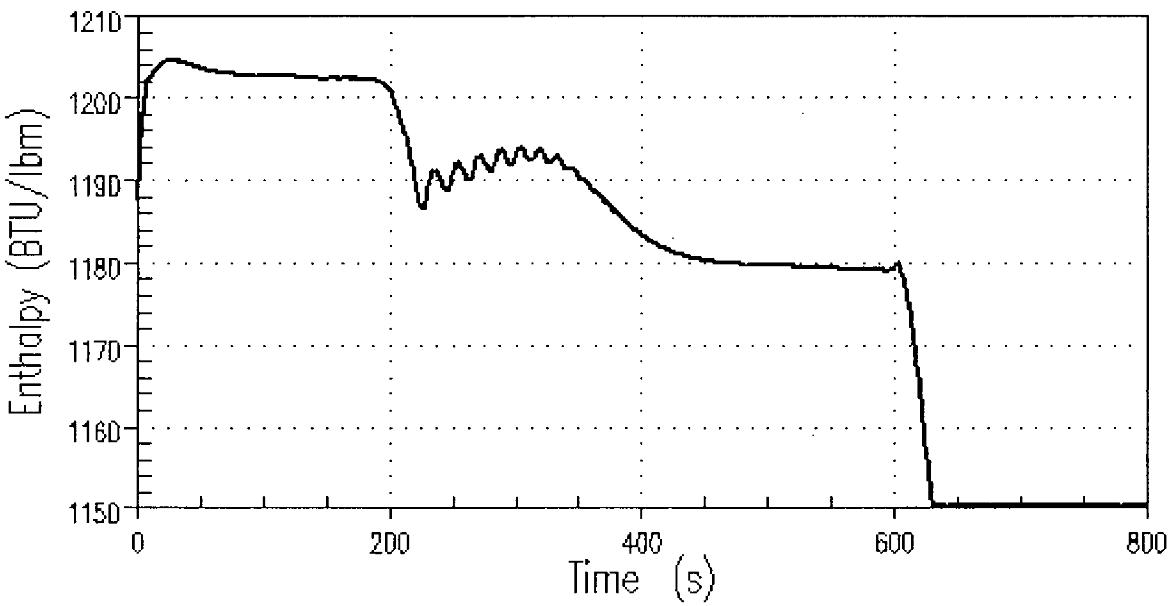

Figure 2.6.3.2-2 **Steam Line Break Enthalpy of Break Effluent 4.7 ft^2 Split Break, 30% Power, Containment Safeguards, and FIV Failure**

2.6.4 Combustible Gas Control in Containment

2.6.4.1 Regulatory Evaluation

Following a loss-of-coolant accident (LOCA), hydrogen and oxygen may accumulate inside the containment due to chemical reactions between the fuel rod cladding and steam, corrosion of aluminum and other materials, and radiolytic decomposition of water. If excess hydrogen is generated, it may form a combustible mixture in the containment atmosphere. The Luminant Power review covered:

- The capability to prevent high concentrations of combustible gases in local areas
- The capability to monitor combustible gas concentrations

The review primarily focused on any impact that the proposed stretch power uprate (SPU) may have on hydrogen mixing.

The acceptance criteria for combustible gas control in containment are based on:

- 10 CFR 50.44, insofar as it requires that certain plants be provided with the capability for controlling combustible gas concentrations in the containment atmosphere.
- General Design Criterion (GDC)-5, insofar as it requires that structures, systems, and components (SSCs) important-to-safety not be shared among nuclear power plants unless it can be shown that sharing will not significantly impair their ability to perform their safety functions.
- GDC-41, insofar as it requires that systems be provided to control the concentration of hydrogen or oxygen that may be released into the reactor containment following postulated accidents to ensure that containment integrity is maintained.
- GDC-42, insofar as it requires that systems required by GDC-41 be designed to permit periodic inspections.
- GDC-43, insofar as it requires that systems required by GDC-41 be designed to permit appropriate periodic testing.

Current Licensing Basis

Hydrogen mixing is described in Final Safety Analysis Report (FSAR) subsection 6.2.5.

On September 16, 2003, the Nuclear Regulatory Commission (NRC) amended 10 CFR 50.44 to eliminate certain requirements for hydrogen recombiners and hydrogen purge systems and relaxed the requirements for hydrogen and oxygen monitoring equipment to make them commensurate with risk significance.

Subsequently, Luminant Power requested and received NRC approval for removal of the electrical hydrogen recombiners from the Technical Specifications (References 1 and 2).

The NRC-approved Technical Specification changes eliminated the need for hydrogen recombiners at CPNPP and conformance to GDCs-5, -41, -42, and -43, with respect to the containment combustible gas control system.

2.6.4.2 Technical Evaluation

Luminant Power has verified that the Comanche Peak Nuclear Power Plant (CPNPP) has a hydrogen monitoring system capable of diagnosing beyond design-basis accidents. In addition, Luminant Power has committed to maintain the hydrogen monitors capable of diagnosing beyond design basis accidents (FSAR Section 7.5).

2.6.4.3 Conclusion

Luminant Power concludes that, based on the license amendment approved by the NRC, the containment combustible gas control system and its components are no longer classified as engineered safety features or safety-related. Hydrogen monitoring will be maintained as described in Reference 1. Therefore, Luminant Power finds the proposed SPU acceptable with respect to combustible gas control in containment.

2.6.4.4 References

1. TXU Power LAR TXX 04167, dated 28 October 2004.
2. LA No. 117, transmitted via NRC Letter, dated 21 April 2005.

2.6.5 Containment Heat Removal

2.6.5.1 Containment Spray System

2.6.5.1.1 Regulatory Evaluation

The containment spray (CT) system removes heat from the containment environment following a loss-of-coolant accident (LOCA), main steam line break (MSLB) accident, or feedwater line break (FWLB) accident. The review in this area focused on the effects of the stretch power uprate (SPU) on the analysis of the available net positive suction head (NPSH) to the CT pumps and the analysis of the heat removal capabilities of the CT heat exchangers.

The acceptance criterion for containment heat removal in conjunction with the CT system is based on General Design Criterion (GDC)-38, insofar as it requires that the CT system be capable of rapidly reducing the containment pressure and temperature following a LOCA, and maintaining them at acceptably low levels.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

GDC-38, detailed in FSAR Section 3.1.4.9, Criterion 38 – Containment Heat Removal, describes two separate CT recirculation trains each with 100-percent capacity, designed to remove heat from the Containment Building following certain design basis accidents (heat removal by the CT system is in conjunction with the emergency core cooling system (ECCS)). Each system is fed from its individual electrical 1E bus, and each bus is connected to a separate offsite power source and is also connected to its individual onsite power source. The system is designed to ensure that the failure of any single active component, assuming the availability of either onsite or offsite power exclusively, does not prevent the system from accomplishing its planned safety function.

The CT system is described in detail in FSAR Section 6.2.2. Other FSAR sections that address the design features and functions of the CT system include:

- FSAR Section 3.5.1.1, Internally Generated Missiles (Outside Containment), which describes the design basis for those systems, including the CT system (as identified by FSAR Table 3.5-1), which require protection from missiles generated outside containment but internal to the plant.

- FSAR Section 3.2.1.1, Seismic Category I, which describes the seismic classification applicable to the CT system in accordance with the seismic requirements of Appendix A to 10 CFR Part 50 (that is, systems that must remain functional during the safe shutdown earthquake are designated Seismic Category I).

- FSAR Section 6.2.4, Containment Isolation System, which describes the design features provided for containment isolation in conjunction with the CT system penetrations.

- FSAR Section 6.5.2, Containment Spray Systems, which describes the design features associated with the chemical additive subsystem of the CT system and the spray header locations and volumetric spray coverage.

2.6.5.1.2 Technical Evaluation

2.6.5.1.2.1 Introduction

The CT system is a safety-related system designed to remove heat from the containment environment following a LOCA, an MSLB, or an FWLB. Major components of the CT system are the refueling water storage tank (RWST), CT pumps, CT heat exchangers, spray headers, spray nozzles, and containment recirculation sumps.

The CT system, in conjunction with the ECCS, removes post-accident thermal energy from the containment environment, thereby reducing Containment pressure and temperature (that is, below containment design limits). The systems must also be capable of limiting post-accident containment conditions such that environmental qualification (EQ) acceptance limits at the SPU program conditions are met.

As described above, the CT system is an active post-accident safety system required to remove heat from containment atmosphere to control containment temperature and prevent over-pressurization of the Containment Building, and to remove containment iodine (See Licensing Report (LR) Section 2.7, Habitability, Filtration, and Ventilation). Each unit of the CPNPP system is equipped with two redundant and physically separated CT trains. Each train draws spray water from a common RWST during the initial injection phase and from separate recirculation sumps during the recirculation phase. Heat is removed from the containment atmosphere by heat transfer to the spray droplets.

Relative to the single failure criterion with respect to the LOCA event, the failure of any single active component during the injection or recirculation phase or failure of a single passive component during the recirculation phase does not prevent safe operation of the system.

2.6.5.1.2.2 Description of Analyses and Evaluation

The CT system and components were evaluated to ensure they are capable of performing their intended functions at SPU conditions. The evaluations compared the existing design parameters of the system/components with the SPU conditions for the following design aspects:

- Containment spray pump performance under SPU operation

- Containment sump pH

- Containment spray heat exchanger performance at the increased SPU heat loads during accident conditions

- Containment spray system temperature limits

- Design pressure/temperature of piping and components versus the SPU operating pressures and temperatures

- Protection of isolated piping sections from heatup effects

Other related evaluations of the containment spray system and components are addressed in the following Licensing Report (LR) sections:

- Piping/component supports – LR subsection 2.2.2.2, BOP – Piping, Components and Supports (Non-Class 1)

- Safety-related valve and pump testing and valve closure, including containment isolation requirements – LR subsection 2.2.4, Safety Related Valves and Pumps

- Post-accident heat removal requirements – LR subsection 2.6.1, Primary Containment Functional Design

2.6.5.1.2.3 Results

The following subsections evaluate the specific containment spray system and component licensing, design, and performance capabilities while at SPU conditions.

General Design Criteria

The evaluation of the containment spray system capabilities at SPU conditions demonstrates that the CPNPP will continue to meet the current licensing basis with respect to the requirements of GDC-38. The system is protected from the dynamic effects of pipe break as described in LR subsection 2.2.1, Pipe Rupture Locations and Associated Dynamic Effects, and LR subsection 2.5.1.3, Pipe Failures. Safety-related equipment is environmentally qualified for the worst case environments as discussed in LR subsection 2.3.1, Environmental Qualification of Electrical Equipment. As described in LR subsection 2.5.1.3, Pipe Failures, the flooding analysis has evaluated the worst case failure in each plant building/area. The containment spray system previously analyzed failure effects are not affected by SPU conditions since the CT system flow rate and pressure does not change at SPU and no physical changes are being made.

Containment Spray Pump Performance

The CT pump NPSH analyses at current power level evaluates the adequacy of the CT system design, both with respect to NPSH margin and with respect to potential fluid flashing in the suction lines during the injection and recirculation phases.

During the injection phase, the NPSH is calculated using the atmospheric pressure in the RWST, the static head between the suction sparger and the pump elevation, the piping losses, and the vapor pressure of water at 120°F.

During the recirculation phase, adequate NPSH for the CT pump is ensured by the design of CT system in accordance with Nuclear Regulatory Commission (NRC) Regulatory Guide 1.1. The static head considers only the difference between the elevation of the pump impeller centerline and the minimum calculated containment water level during the recirculation phase.

There are no changes to CT pump flow, containment sump water level, or pump NPSH available as a result of SPU.

Containment Sump pH

The major attributes relative to CT sump pH are the delivered volumes to the sump and the buffering solution concentrations. There are no changes to CT sump pH as a result of SPU.

Containment Spray System Heat Removal Capability

CT heat exchangers are cooled by component cooling (CC) water and are located in the CC safeguards loop. The SPU increases heat available to be released into containment, and thus subsequent heat loads on the containment heat removal systems. The required heat removal by the CT system during and after LOCA reflects a slight increase. Thermal performance of the CT system under SPU conditions was analyzed for the scenario when the system is rejecting the required heat loads under accident conditions (LOCA). Under SPU operation, CT heat exchangers as designed are capable of removing required heat loads during LOCA.

Relative to safety function of the CT system (in conjunction with the ECCS system), LR subsection 2.6.1, Primary Containment Function Design, shows continued compliance with acceptance limits.

SPU Operating Conditions Versus Design Conditions of Piping and Components

The CT system flow rates are unchanged under SPU conditions. No physical modifications are required to be made to the system (that is, no piping modifications and no pump modifications). Operating design pressures are unchanged. There fore, since the containment spray system operating design pressures are not affected by SPU operation, the existing piping and component design pressures continue to bound operating pressures and are acceptable for SPU operation.

Design temperature of the CT pumps and the CT heat exchangers (tube side) is 300°F. These design temperatures as well as design temperature of CT piping bound the maximum component spray water operating temperatures at SPU conditions.

NRC Generic Letter 96-06

The CT heat exchangers use component cooling water as a cooling medium to remove heat from containment. The issue in NRC Generic Letter 96-06 related to the heatup/ overpressurization of isolated component cooling water piping inside containment and was evaluated by Luminant Power in previous responses to the NRC, as detailed in LR subsection 2.5.4.3.

NRC Generic Letter 2004-02

The issue in NRC Generic Letter 2004-02 relates to the potential impact of debris blockage on emergency recirculation during design basis accidents. Specifically, in conjunction with GSI-191, the objective is to ensure that post-accident debris blockage will not impede or prevent operation of the ECCS and CT system in the recirculation mode during LOCAs or other high energy line break (HELB) accidents for which sump recirculation is required.

The modifications and actions required by NRC GL 2004-02 are not affected by SPU. The impact of associated GSI-191 modifications will be addressed as part of the final response to GL 2004-02.

2.6.5.1.3 Conclusion

The Luminant Power review has assessed the effects of the proposed SPU on the CT system and has adequately accounted for the increased heat loads from the proposed SPU on system performance. The review concluded that the CT system will provide adequate containment heat removal capability during and after LOCA conditions to reduce containment pressure and temperature and maintain these parameters at acceptably low levels. Therefore, it is concluded that the CT system will meet the current licensing basis with respect to the requirements of GDC-38. Based on the above, the proposed SPU is acceptable with respect to the CT system.

2.6.6 Pressure Analysis for ECCS Performance Capability

2.6.6.1 Regulatory Evaluation

Following a loss-of-coolant-accident (LOCA), the emergency core cooling system (ECCS) will supply water to the reactor vessel to reflood, and thereby cool the reactor core. The core flooding rate will increase with increasing containment pressure. Luminant Power reviewed analyses of the minimum containment pressure that could exist during the period of time until the core is reflooded to confirm the validity of the containment pressure used in ECCS performance capability studies. The review covered assumptions made regarding heat removal systems, structural heat sinks, and other heat removal processes that have the potential to reduce the pressure. The acceptance criteria for the pressure analysis for ECCS performance capability are based on 10 CFR 50.46, insofar as it requires the use of an acceptable ECCS evaluation model that realistically describes the behavior of the reactor during LOCAs or an ECCS evaluation model developed in conformance with 10 CFR Part 50, Appendix K.

2.6.6.2 Technical Evaluation

This is included in the large break LOCA (LBLOCA) submittal (TXX-07107).

2.6.6.3 Conclusion

Luminant Power has reviewed the impact that the proposed stretch power uprate (SPU) would have on the minimum containment pressure analysis and concludes that the requirements in 10 CFR 50.46 regarding ECCS performance will continue to be met following implementation of the proposed SPU. Therefore, Luminant Power finds the proposed SPU acceptable with respect to minimum containment pressure for ECCS performance.

2.7 HABITABILITY, FILTRATION, AND VENTILATION

2.7.1 Control Room Habitability System

2.7.1.1 Regulatory Evaluation

The safety function of the Comanche Peak Nuclear Power Plant (CPNPP) Control Room habitability system is to ensure that plant operators are adequately protected from the effects of accidental releases of toxic and radioactive gases both during normal operation and for a period of not less than 30 days after a loss-of-coolant accident (LOCA). The protected envelope includes the Control Rooms, equipment rooms, instrument rooms, computer rooms, offices, electrical equipment corridor, kitchen, sanitary facilities, and the Technical Support Center. Luminant Power review focused on the effects of the proposed stretch power uprate (SPU) on radiation doses, toxic gas concentrations, and estimates of dispersion of airborne contamination.

The acceptance criteria for the Control Room habitability system are based on:

- General Design Criterion (GDC)-4, insofar as it requires that structures, systems, and components (SSC) important to safety be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with postulated accidents, including the effects of the release of toxic gases.
- GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the Control Room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent, to any part of the body, for the duration of the accident.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the adequacy of the CPNPP Control Room habitability system is assessed relative to conformance with the following:

- GDC-4 is described in FSAR Section 3.1.1.4, General Design Criteria 4 – Environmental and Dynamic Effects Design Basis. Conformance to the requirements of GDC-4 is described in the following:
  - Environmental Design of Mechanical and Electrical Equipment (FSAR Section 3.11)

- Protection Against the Dynamic Effects Associated With the Postulated Rupture of Piping (FSAR Section 3.6)
- Missile Protection (FSAR Section 3.5)

- GDC-19 is addressed in FSAR Section 3.1.2.10. CPNPP is equipped with a Control Room that contains controls and instrumentation as necessary for operation of the reactor and turbine generator under normal and accident conditions. The Control Room is capable of continuous occupancy by operating personnel under all operating and accident conditions, within specified dose limits.

2.7.1.2 Technical Evaluation

The functions of the Control Room habitability system are performed by the Control Room heating, ventilation, and air conditioning (HVAC) system. The Control Room HVAC system has four different operating modes that allow it to provide a safe atmosphere in the Control Building envelope during all postulated accident conditions. During loss-of-offsite power, safety injection, or high radiation conditions, the HVAC system goes into the emergency recirculation mode. In this mode, the normal makeup fans and exhaust fans are stopped, and the air conditioning units, the emergency recirculation filtration units, and the emergency pressurization filter units are all automatically started.

On detection of smoke at the fresh air intake, the system is manually aligned to other intake or to the isolation mode. In the isolation mode the outdoor intake dampers are shut, the makeup air supply fans and the exhaust are shut down, and the emergency filtration units are started. Further discussion of the HVAC system can be found in Licensing Report (LR) subsection 2.7.3.1.

The HVAC system is designed to protect the Control Room from the effects of external events including seismic, weather, radiation, and smoke.

The radiological consequences to the Control Room remain within regulatory limits as discussed in LR Section 2.9. There is an increase in dose for most events except for the LOCA condition. Since the Control Room will see the highest impact from the LOCA, and the dose does not increase for this condition, it is concluded that the radiological consequences to the Control Room will not be impacted by the uprate. The Technical Support Center is located in the Control Building and is contained within the Control Room habitability envelope.

The operation of the Control Room habitability system is not affected by the SPU operation. Likewise, there are no new chemicals of combustible materials stored near or on-site as a result of the SPU. There are no modifications or additions to system components as the result of SPU that would introduce any new functions or change the functions of existing components that would affect the evaluation of existing system boundaries. Operation of the Control Room habitability system at SPU conditions does not add any new types of materials or previously unevaluated materials to the system. System component internal and external environments remain within the parameters previously evaluated.

2.7.1.3 Conclusion

Luminant Power has reviewed the effects of the proposed SPU on the ability of the Control Room habitability system to protect plant operators from the effects of accidental releases of radiological gases. Luminant Power has adequately accounted for the increase of radioactive gases that would result from the proposed SPU and concludes that the Control Room habitability system will continue to provide the required protection following implementation of the proposed SPU. Based on this, the Control Room habitability system will continue to meet the requirements of GDCs-4 and -19. Therefore, the proposed SPU is acceptable with respect to the Control Room habitability system.

2.7.2 Engineered Safety Feature Atmosphere Cleanup

2.7.2.1 Regulatory Evaluation

The engineered safety features (ESF) atmosphere cleanup systems are designed for fission product removal in post-accident environments. These systems at the Comanche Peak Nuclear Power Plant (CPNPP) include the containment spray system, the emergency filtration and pressurization units in the Control Room area ventilation system, and the primary plant ventilation ESF exhaust units. For each ESF atmosphere cleanup system, the review focused on the effects of the proposed stretch power uprate (SPU) on system functional design, environmental design, and provisions to preclude temperatures in the absorber section from exceeding design limits.

The acceptance criteria for the ESF atmosphere cleanup systems are based on:

- General Design Criterion (GDC)-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the Control Room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent, to any part of the body, for the duration of the accident.
- GDC-41, insofar as it requires that systems to control fission products released into the reactor containment be provided to reduce the concentration and quality of fission products released to the environment following postulated accidents.
- GDC-61, insofar as it requires that systems that may contain radioactivity be designed to assure adequate safety under normal and postulated accident conditions.
- GDC-64, insofar as it requires that means shall be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences (AOOs) and postulated accidents.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

The adequacy of CPNPP design relative to conformance to:

- GDC-19 is described in the FSAR Section 3.1.2.10, Control Room (Criterion 19). As described in this FSAR section, a Control Room shall be provided with adequate radiation protection to permit access and occupancy of the Control Room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

- Safe occupancy of the Control Room during abnormal conditions is provided for in the design. The Control Room is in a seismic Category I structure. Adequate shielding is provided for the Control Room in the event of a design basis accident (DBA). The Control Room air conditioning system features redundant equipment, and radiation and smoke detectors, with appropriate alarms and interlocks. Provisions are made for Control Room air to be recirculated through high-efficiency particulate air (HEPA) and charcoal filter.

- GDC-41 is described in FSAR Section 3.1.4.12, Containment Atmosphere Cleanup. This FSAR section describes the containment spray system, including the sodium hydroxide injection system that is designed to provide fission product cleanup of the containment atmosphere after a postulated loss-of-coolant accident (LOCA). The design and operation of the containment spray and sodium hydroxide injection systems are described in FSAR Section 6.5.2.

- GDC-61 is described in FSAR Section 3.1.6.2, Fuel Storage and Handling and Radioactivity Control. As described in this FSAR section, the spent fuel pool storage and handling system is designed to provide adequate cooling, containment, shielding, and filtration to ensure safety during all normal and postulated accident conditions. Control of airborne radioactivity is further described in FSAR Section 9.4.4 and Licensing Report (LR) subsection 2.7.4.

- GDC-64 is described in FSAR Section 3.1.6.5, Monitoring Radioactive Releases. The containment atmosphere is continuously monitored during normal and transient operations using the containment particulate and gas monitors. Radioactivity levels contained in the facility effluent discharge paths and in the environs are continually monitored during normal and accident conditions by the station radiation monitoring system and by the Radiation Protection Program described in FSAR Sections 11.5 and 12.5.

2.7.2.2 Technical Evaluation

Control Room – The Control Room area ventilation system includes two sets of HEPA/carbon filters for use during accident conditions, the emergency filtration units, and the emergency pressurization units. These two sets of filters automatically provide clean air for pressurization of the Control Room envelope during accident conditions, and recirculation of the Control Room atmosphere to provide additional assurance that the dose levels are within requirements. The dose analysis for SPU operation presented in LR Section 2.9 indicated that existing system and equipment are capable of maintaining the Control Room envelope dose within regulatory limits. Additional information on the Control Room ventilation system can be found in LR subsections 2.7.1 and 2.7.3.

Containment – The containment spray system, along with the sodium hydroxide injection, is designed to remove heat and fission products from the containment atmosphere after a LOCA. The system includes redundant, Category I spray trains that permanently remove iodine from the containment atmosphere by absorption into the droplets and retention in the containment

sump. The SPU will increase the core radioactive source, but the offsite, and Control Room dose analyses presented in LR Section 2.9 demonstrate the ability of the existing containment spray system to maintain the releases within acceptable limits.

Primary plant ventilation – The primary plant ventilation exhaust system includes four ESF exhaust units that maintain the Auxiliary, Fuel Handling, and Safeguards Buildings at a negative pressure during LOCA conditions. Each exhaust unit has a dedicated fan, HEPA filters, and carbon absorbers to prevent the release of radioactive contamination into the atmosphere. Maintaining a negative pressure in the buildings will prevent exfiltration of any contamination. The ESF exhaust units are safety Class 3, seismic Category I, and 100-percent redundant. The results of the dose analysis in LR Section 2.9 indicate that the SPU results in no increase in the duty of the ESF exhaust units.

Results

The SPU does not require modifications to the equipment or operation of any of the engineered safety features atmospheric cleanup systems.

2.7.2.3 Conclusion

Luminant Power review of the effects of the proposed SPU on the ESF atmosphere cleanup systems concludes that the design is adequate for the increase of fission products and changes in expected environmental conditions that would result from the proposed SPU. The ESF atmosphere cleanup systems will continue to provide adequate fission product removal in post-accident environments following implementation of the proposed SPU. Based on this, the ESF atmosphere cleanup systems will continue to meet the requirements of GDCs-19, -41, -61, and -64.

2.7.3 Control Room Area Ventilation Systems

2.7.3.1 Regulatory Evaluation

The function of the Control Room area ventilation system (Control building ventilation system) is to provide a controlled environment for the comfort and safety of Control Room personnel and to support the operability of Control Room components during normal operation and design basis accident (DBA) conditions. The Luminant Power review of the Control Building ventilation system focused on the effects that the proposed stretch power uprate (SPU) will have on the functional performance of safety-related portions of the system. The Luminant Power review included the effects of radiation, combustion, and other toxic products; and the expected environmental conditions in areas served by the Control Building ventilation system.

The acceptance criteria for the review are:

- General Design Criterion (GDC)-4, insofar as it requires that structures, systems, and components (SSC) important to safety be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.

- GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the Control Room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

- GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

The adequacy of CPNPP design relative to conformance to:

- GDC-4 is described in FSAR Section 3.1.1.4, Environmental and Missile Design Bases (Criterion 4).

  As described in this FSAR section, SSCs important to safety are designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operating, maintenance, testing, and postulated accidents including loss-of-coolant accidents (LOCAs). These items are either protected from accident conditions or designed to withstand, without failure, exposure to the

combination of temperature, pressure, humidity, radiation, and dynamic effects expected during the required operational period.

Physical separation, physical protection, pipe restraints, and redundancy are included in the design of safety-related systems to ensure that each such system performs its intended safety function.

SSCs important to safety are classified as Seismic Category I and are designed in accordance with the codes and classifications indicated in FSAR Section 3.2.1.1.

- GDC-19 is described in FSAR Section 3.1.2.10, Control Room (Criterion 19). As described in this FSAR section, the Control Room is equipped to operate the unit safely under normal and accident conditions. Its shielding and ventilation design permits continuous occupancy of the Control Room for the duration of a DBA without the dose to personnel exceeding 5 rem whole body.

- GDC-60 is described in FSAR Section 3.1.6.1, Control of Releases of Radioactive Materials to the Environment (Criterion 60).

  In all cases, the design for radioactivity control is based on:

  1. The requirements of 10 CFR 20, 10 CFR 50, and Appendix I to 10 CFR 50 for normal operations and for transient situation that might reasonably be anticipated to occur.

  2. 10 CFR 100 dose level guidelines for potential accidents of extremely low probability of occurrence.

All release paths, including ventilation and process streams, are monitored and controlled as described in FSAR Section 11.5.

2.7.3.2 Technical Evaluation

2.7.3.2.1 Introduction

The Control Room area ventilation system includes the following major components:

- Control Room area air conditioning units
- Control Room toilet and kitchen exhaust ventilation fans
- Control Room normal exhaust fans
- Control Room emergency filtration units
- Control Room emergency pressurization units
- Control Room normal supply fans

The Seismic Category 1 Control Room air conditioning subsystem is designed to provide the Control Room habitability envelope with cooled air during both normal and accident conditions.

The Control Room habitability envelope is maintained at a positive pressure relative to the other portions of the Control Building and relative to the outside atmosphere before, during, and after a postulated accident. The positive pressure is achieved by mechanically introducing outside air through the air handling units during normal operation, and through the emergency pressurization filters during accident conditions. The positive pressure prevents inleakage of radioactive air and satisfies the radiation exposure limit criteria of GDC-19.

2.7.3.2.2 Description of Analyses and Evaluations

The Control Building ventilation system was evaluated to ensure it is capable of performing its intended functions at SPU conditions. The design and operation of the Control Room ventilation system are unaffected by the SPU. Smoke, toxic gas, and external event assumptions and conclusions are also unaffected by the SPU.

Other evaluations are addressed in the following Licensing Report (LR) sections:

- Control room habitability – LR subsection 2.7.1, Control Room Habitability System
- Radiological analysis methods and assumptions – LR Section 2.9, Source Terms and Radiological Consequences
- Control of the release of radioactive effluents – LR subsection 2.10.1, Occupational and Public Radiation Doses

2.7.3.2.3 Results

The proposed SPU has no effect on the ability of the Control Room ventilation system to provide a controlled environment for the comfort and safety of Control Room personnel and to support the operability of Control Room components. CPNPP has adequately accounted for the increase of radioactive gases that would result from a DBA under the proposed SPU operating conditions, and any associated changes to parameters affecting environmental conditions for Control Room personnel and equipment. The Control Building ventilation system will continue to provide an acceptable Control Room environment for safe operation of the plant following implementation of the proposed SPU. The design criteria, design bases, and safety classification for the Control Building ventilation system, and the requirements for system performance continue to provide conformance with the requirements of GDCs-4, -19, and -60. Based on this, the Control Building ventilation system will continue to meet the requirements of GDCs-4, -19, and -60 for SPU operating conditions. Therefore, the proposed SPU is acceptable with respect to the Control Building ventilation system.

2.7.3.3 Conclusion

The effects of the proposed SPU on the ability of the Control Building ventilation system to provide a controlled environment for the comfort and safety of Control Room personnel and to support the operability of Control Room components were reviewed. CPNPP has adequately accounted for the increase of radioactive gases that would result from a DBA under the conditions of the proposed SPU, and associated changes to parameters affecting environmental conditions for Control Room personnel and equipment. The Control Building ventilation system will continue to provide an acceptable Control Room environment for safe operation of the plant following implementation of the proposed SPU. Based on this, the Control Room ventilation system will continue to meet the CPNPP current licensing basis with respect to the requirements of GDCs-4, -19, and -60. Therefore, the proposed SPU is acceptable with respect to the Control Building ventilation system.

2.7.4 Spent Fuel Pool Area Ventilation System

2.7.4.1 Regulatory Evaluation

The function of the spent fuel pool area ventilation system is to maintain ventilation in the spent fuel pool areas, to permit personnel access, and to control airborne radioactivity in the area during normal operation, anticipated operational occurrences, and following postulated accidents. The Luminant Power review focused on the effects of the proposed stretch power uprate (SPU) on the functional performance of the safety-related portions of the system. The acceptance criteria for the spent fuel pool area ventilation system are based on:

- General Design Criterion (GDC)-60, insofar as it requires that the plant design includes means to control the release of radioactive effluents.

- GDC-61, insofar as it requires that systems containing radioactivity be designed with appropriate confinement and containment.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Conformance to GDCs -60 and -61 is addressed in FSAR Sections 3.1.6.1 and 3.1.6.2, respectively. GDC-60 requires that the plant design include means to suitably control the release of radioactive materials in gaseous effluents and that sufficient holdup capacity be provided for retention of gaseous effluents containing radioactive materials. FSAR Section 3.1.6.1 states that the handling, control, and release of radioactive materials during Modes 1 and 2 is in compliance with 10 CFR 50, Appendix I, and is described in the Offsite Dose Calculation Manual.

The requirements of GDC-61 relevant to the spent fuel pool ventilation system and gaseous effluents treatment system require that the ventilation system be designed to assure adequate safety under normal and postulated accident conditions. The system shall be designed (1) to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, and (3) with appropriate containment, confinement, and filtering systems. FSAR Section 3.1.6.2 states that the gaseous waste management system is designed to ensure adequate safety under normal and postulated accident conditions by providing the following:

- Components are designed and located such that appropriate periodic inspection and testing may be performed.

- All areas of the plant are designed with suitable shielding for radiation protection based on anticipated radiation dose rates and occupancy as discussed in FSAR Chapter 12.

- Individual components that contain significant radioactivity are located in confined areas that are adequately ventilated through appropriate filtering systems.

The function of the spent fuel pool area ventilation system portion of the Fuel Building ventilation system is addressed in primary plant ventilation system Technical Specification bases B3.7.12.

2.7.4.2 Technical Evaluation

2.7.4.2.1 Introduction

The spent fuel pool area ventilation system is part of Fuel Building ventilation system. The Fuel Building ventilation system is described in FSAR Section 9.4.2. The impact of the proposed SPU on the fuel building ventilation system is further evaluated in Licensing Report (LR) subsection 2.7.6, Engineered Safety Features Ventilation System. As described in the FSAR, the spent fuel pool ventilation system serves to maintain required air temperature and to control airborne radioactivity in the spent fuel pool area during normal operating conditions. This is accomplished by directing air from the primary plant supply air units from above ductwork to the spent fuel pool and to exhaust air around the perimeter of the pools with ducts connected to the suction of the primary plant modular exhaust filtration units. Exhaust air from the spent fuel pool water surface is drawn through roughing filters, high efficiency particulate air (HEPA) filters and charcoal filters. Discharge from the primary plant modular exhaust filtration units main exhaust fans is then routed out to the plant ventilation discharge vent to atmosphere.

2.7.4.2.2 Description of Analyses and Evaluations

The spent fuel pool area ventilation system was evaluated to ensure it is capable of performing its intended functions at SPU conditions. The decay heat loads in the spent fuel pool will increase due to the SPU conditions, but pool water temperatures will remain below pre-SPU design limits. SPU decay heat loads and pool water temperatures have been evaluated to ensure that the system is capable of performing its intended functions under normal SPU and refueling modes. Other evaluations are addressed in the following LR sections:

- Spent fuel pool cooling system evaluation – LR subsection 2.5.4.1, Spent Fuel Pool Cooling and Cleanup System

- Off-site dose consequences of a fuel handling accident – LR subsection 2.9.8, Radiological Consequences of Fuel Handling Accident

- Control of the release of radioactive effluents – LR subsection 2.10.1, Occupational and Public Radiation Doses

2.7.4.2.3 Results

The temperature in the spent fuel pool area is a function of the heat released from the spent fuel pool. The decay heat in the spent fuel is slightly greater at SPU conditions, but the spent fuel pool water temperature during normal and abnormal SPU operation remains below design

conditions. Refer to LR subsection 2.5.4.1, Spent Fuel Pool Cooling and Cleanup System. Therefore, the spent fuel pool area ventilation system will maintain the required temperature conditions for personnel and equipment during SPU operation.

Since the design of the spent fuel pool area ventilation system has not changed following the implementation of the SPU, airborne radioactivity released from the spent fuel in the pool will continue to be collected and exhausted by the primary plant ventilation system after passing through roughing filters, HEPA filters, and charcoal filters. Therefore, the control of airborne radioactivity in the spent fuel pool area is not affected following implementation of the SPU.

The evaluation of the spent fuel pool area ventilation system at SPU conditions demonstrates that the CPNPP will continue to meet the current licensing basis with respect to GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents. This system was evaluated in LR subsection 2.10.1, Occupational and Public Radiation Doses, and no changes are required as a result of the SPU. The handling, control, and release of radioactive materials are in compliance with 10 CFR 50, Appendix I, and is described in the Off-site Dose Calculation Manual.

The evaluation of the spent fuel pool area ventilation system at SPU conditions demonstrates that the CPNPP will continue to meet the current licensing basis with respect to GDC-61, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement to ensure adequate safety under normal and postulated accident conditions. This design capability remains unchanged by the SPU.

The evaluation of the ability of the spent fuel pool area ventilation system to maintain the required temperature conditions and to contain radioactivity to permit personnel access during the SPU demonstrates that there is no effect on this system design capability by the SPU.

SPU activities do not add any new components nor do they introduce any new functions for existing components that would change the licensed system boundaries. There are no changes associated with operation of the spent fuel pool ventilation system at SPU conditions and the SPU does not add any new or previously unevaluated materials to the system. System component internal and external environments remain within the parameters previously evaluated.

2.7.4.3 Conclusions

The Luminant Power review determined that the spent fuel pool area ventilation system has adequately accounted for the effects of the proposed SPU on the system's capability to maintain required ambient conditions in the spent fuel pool and equipment areas, permit personnel access, control airborne radioactivity in the area, control release of gaseous radioactive effluents to the environment, and provide appropriate containment. Based on this, the Luminant Power review concluded that the spent fuel pool ventilation system will continue to meet the CPNPP current licensing basis with respect to the requirements of GDCs-60 and -61, and 10 CFR 50, Appendix I. Therefore, the proposed SPU is acceptable with respect to the spent fuel pool areas ventilation system.

2.7.5 Auxiliary and Radwaste Area and Turbine Area Ventilation Systems

2.7.5.1 Regulatory Evaluation

The function of the auxiliary and radwaste area ventilation system and the turbine area ventilation system is to maintain ambient temperatures in the auxiliary and radwaste equipment and turbine areas, permit personnel access, and control the concentration of airborne radioactive material in these areas during normal operation, during anticipated operational occurrences, and after postulated accidents. The Luminant Power review focused on the effects of the proposed stretch power uprate (SPU) on the functional performance of the safety-related portions of these systems. The acceptance criteria for the auxiliary and radwaste area ventilation system and turbine area ventilation system are based on General Design Criterion (GDC)-60, insofar as it requires that the plant design includes means to control the release of radioactive effluents.

The ventilation systems important to personnel safety or vital equipment operation are addressed in Licensing Report (LR) subsection 2.7.6, Engineered Safety Feature Ventilation System.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Sections 3.1.1 and 3.1.6.

Conformance to GDC-60 is addressed in FSAR Section 3.1.6.1. GDC-60 requires that the plant design include means to suitably control the release of radioactive materials in gaseous effluents and that sufficient holdup capacity be provided for retention of gaseous effluents containing radioactive materials. FSAR Section 3.1.6.1 states that the handling, control, and release of radioactive materials during Modes 1 and 2 is in compliance with 10 CFR 50, Appendix I, and is described in the Off-site Dose Calculation Manual.

In all cases, the design for radioactivity control is based on:

- The requirements of 10 CFR 20, 10 CFR 50, and Appendix I to 10 CFR 50 for normal operations and for transient situation that might reasonably be anticipated to occur.
- 10 CFR 100 dose level guidelines for potential accidents of extremely low probability of occurrence.

All release paths, including ventilation and process streams, are monitored and controlled as described in FSAR Section 11.5.

2.7.5.2 Technical Evaluation

2.7.5.2.1 Introduction

The auxiliary and radwaste area ventilation system (part of primary plant ventilation) and the turbine area ventilation system provide heating, ventilation, and air conditioning (HVAC) to vital and non-vital areas and plant equipment. The principle components of the these ventilation system include filters, fans, dampers, valves, heat exchangers, conditioning/chiller packages, and the ductwork, piping, and valves.

The Turbine Building ventilation system is discussed in FSAR Section 9.4.4. The Turbine Building ventilation system uses roof vent fans, wall vent fans, louvers, and electric unit heaters for ventilation and temperature control. Included in the turbine building ventilation service are the main feedwater pumps, the condensate pumps, and heater drain pumps areas.

The Auxiliary Building ventilation system is designed in accordance with the following criteria:

- GDC-2, for the Auxiliary Building to protect against natural phenomena (Chapters 2 and 3).
- GDC-4, for the Auxiliary Building ventilation ducts and components to protect against adverse environmental conditions and missiles (Chapters 2 and 3)
- GDC-60 and -64 for control and monitoring radioactivity releases in the Auxiliary Building ventilation system.
- GDC-5 for shared systems and components important to safety.
- Regulatory Guides (RG) 1.29, Revision 2 (2/76) for seismic design classification of system components.
- RG 1.52, Revision 1 (7/76) for air filtration and adsorption units.
- Branch Technical Position (BTP) ASB 3-1 and MEB 3-1 for breaks in high and moderate energy piping systems outside containment.
- BTP APCSB 9.5-1, Appendix A, Fire Protection for Nuclear Power Plants.

The radwaste area ventilation system is designed in accordance with the following criteria:

- RG 1.29, Revision 2 (2/76) for seismic design classification of system components. Only those portions of the radwaste ventilation system located inside the Auxiliary Building and designated ANSI Safety Class 3 are classified as seismic Category I and designed to RG 1.75, Revision 1 (1/75), Section 1.8. The safety-related portions of the radwaste exhaust ventilation are located in the Auxiliary Building, a seismic- and tornado-protected

structure. This system is capable of withstanding, or is protected from, the effects of external missiles and internally generated missiles.

- GDC-2, for protect against natural phenomena.

- GDC-5 for sharing structures, systems, and components important to safety.

- GDC-60 and RG 1.140 (3/78) for control of releases of radioactive materials to the environment.

- Areas subject to radioactive contamination are mechanically exhausted with air supplied from primary plant ventilation supply units and natural outdoor or adjacent areas. This maintains a slightly negative pressure relative to the surrounding areas, minimizing the spread of the contamination as required by RG 1.143, Revision 1 (formally BTP ETSB 11-1, Rev. 1) (Section 1.8).

The Turbine Building area ventilation system is designed in accordance with the following criteria:

- The system is non-safety related and is designated non-nuclear safety class.
- BTP APCSB 9.5-1, Appendix A, Fire Protection for Nuclear Power Plants.

As addressed in FSAR Section 9.4.3, Auxiliary and Radwaste Area Ventilation System, the CPNPP Auxiliary Building ventilation system and waste disposal area ventilation system are in conformance with the requirements of GDC-2, -5, and -60 as they relate to protection against natural phenomena, assurance of proper operating environment for essential equipment, shared systems, control of releases of radioactive materials to the environment, and guidelines of RGs 1.29, Revision 2 (2/76) (Positions C.1 and C.2), 1.52, Revision 1 (7/76) (Position C.2), and 1.140 (3/78) (Positions C.1 and C.2) as they relate to seismic classification and system design for emergency and normal operation. The Auxiliary Building ventilation system and radwaste area ventilation system meet the acceptance criteria of Standard Review Plan (SRP) 9.4.3.

As addressed in FSAR Section 9.4.4, Turbine Building Ventilation System, the CPNPP Turbine Building ventilation system meets the requirements of GDC-2 with respect to the need for protection against natural phenomena because its failure does not affect safety system functions or result in release of radioactive material, and meets the guidelines of RG 1.29 Revision 2 (2/76) (Position C.2) concerning its seismic classification. The Turbine Building ventilation system meets the acceptance criteria of SRP 9.4.4.

2.7.5.2.2 Description of Analyses and Evaluations

The changes in heat loads for the ventilation subsystems in areas served by the auxiliary (Auxiliary Building ventilation) and radwaste area ventilation systems and the turbine area (Turbine Building area ventilation) ventilation system were evaluated to ensure that the ventilation systems are capable of performing their intended functions under SPU conditions and found changes to be insignificant to degrade essential system operation, to impact system's

capability to circulate sufficient air to prevent accumulation of flammable or explosive gases, or to impact its ability to control airborne particulate material accumulation.

Other evaluations related to the auxiliary and radwaste area ventilation system and the turbine area (Turbine Building area ventilation) ventilation system are addressed in the following LR sections:

- The engineered safety features (safeguards) building ventilation system and emergency generator enclosure ventilation system are evaluated in LR subsection 2.7.6, Engineered Safety Feature Ventilation System.
- Protection against dynamic effects, including GDC-4 requirements, of missiles, pipe whip, and discharging fluids – LR subsection 2.2.1, Pipe Rupture Locations and Associated Dynamic Effects; and LR subsection 2.5.1.3, Pipe Failures.
- Electrical equipment qualification – LR subsection 2.3.1, Environmental Qualification of Electrical Equipment.
- Onsite and offsite electric power systems, including GDC-17 requirements – LR subsection 2.3.2, Offsite Power System; and LR subsection 2.3.3, AC Onsite Power System.
- Protection against turbine missiles and internal missiles is discussed in LR subsection 2.5.1.2, Missile Protection.

2.7.5.2.3 Results

The auxiliary and radwaste area ventilation systems and the turbine area (Turbine Building area ventilation) ventilation system's ability to provide required temperature conditions for personnel and equipment during normal operation is unaffected by the changes proposed for the SPU. The increase heat loads in these areas are primarily due to changes in the main steam and feedwater system operating conditions, increased brake horsepower for the condensate booster and feedwater pumps, and small increases in electrical loads. For plant areas that use outside air exchange to provide cooling, outside air temperature changes dominate any potential temperature changes caused by the SPU.

The evaluation of the plant equipment changes for the proposed SPU did not identify any need to modify the auxiliary and radwaste area ventilation system and the turbine area (Turbine Building area ventilation) ventilation system. There are no equipment changes as a result of the SPU that could create a new potentially unmonitored radioactive release path. Thus, following the SPU, CPNPP will continue to meet the current licensing basis with respect to GDC-60. The effects of potential releases to the environment are evaluated in LR subsection 2.10.1, Occupational and Public Radiation Doses, and remains within current limits following the SPU. The handling, control, and release of radioactive materials are in compliance with 10 CFR 50, Appendix I, and is described in the Offsite Dose Calculation Manual.

The evaluation of the ability of the auxiliary and radwaste area ventilation system and the turbine area (Turbine Building area ventilation) ventilation system to maintain the required temperature conditions and to contain radioactivity to permit personnel access during the SPU did not identify any need to modify the auxiliary and radwaste area ventilation system and the turbine area (Turbine Building area ventilation) ventilation system and that there is no effect on this system design capability by the SPU.

2.7.5.3 Conclusions

The evaluation of the auxiliary and radwaste area ventilation system and the turbine area (Turbine Building area ventilation) ventilation has adequately accounted for the effects of the proposed SPU on the system's capability to maintain ventilation in the Turbine, Auxiliary, and Radwaste Buildings; permit personnel access, control airborne radioactivity in the area, and control release of gaseous radioactive effluents to the environment. Based on this, Luminant Power concludes that the auxiliary and radwaste area ventilation system and the turbine area (Turbine Building area ventilation) ventilation system will continue to meet the CPNPP current licensing basis with respect to the requirements of GDC-60. Therefore, Luminant Power finds that the proposed SPU is acceptable with respect to the auxiliary and radwaste area ventilation system and the turbine area (Turbine Building area ventilation) ventilation system.

2.7.6 Engineered Safety Feature Ventilation System

2.7.6.1 Regulatory Evaluation

The function of the engineered safety feature(ESF) (safeguards) ventilation system is to provide a suitable and controlled environment for ESF components following certain anticipated transients and design basis accidents. The Luminant Power review for the engineered safety feature ventilation system focused on the effects of the proposed stretch power uprate (SPU) on the functional performance of the safety-related portions of the systems. The Luminant Power review also covered:

- The ability of the ESF equipment in the areas being serviced by the ventilation system to function under degraded ESF ventilation system performance.
- The capability of the ESF ventilation system to circulate sufficient air to prevent accumulation of flammable or explosive gas or fuel-vapor mixtures from components (such as storage batteries and stored fuel).
- The capability of the ESF ventilation system to control airborne particulate material (dust) accumulation.

The Nuclear Regulatory Commission's (NRC's) acceptance criteria for the ESF ventilation system are based on:

- General Design Criterion (GDC)-4, insofar as it requires that structures, systems, and components (SSCs) important to safety be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.
- GDC-17, insofar as it requires onsite and offsite electric power systems be provided to permit functioning of safety-related SSCs.
- GDC-60, insofar as it requires that the plant design includes means to control the release of radioactive effluents.

The Comanche Peak Nuclear Power Plant (CPNPP) Final Safety Analysis Report (FSAR) describes the various plant ventilation systems, primarily in Section 9.4, Air Conditioning, Heating, Cooling and Ventilation Systems. It is noted that the physical location within buildings of functionally important equipment served by the ESF (safeguards) ventilation system (essential ventilation) are areas of the Engineered Safety Features (Safeguards) Building, Auxiliary Building, and radwaste area, Fuel Building, and Diesel Generator Building.

The Control Room emergency air treatment system, Control Building ventilation for areas other then the Control Room, and the Technical Support Center Ventilation systems are discussed in Licensing Report (LR) subsection 2.7.1, Control Room Habitability System.

Other systems are addressed elsewhere as called out in RS-001.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

- GDC-4 is described in FSAR Section 3.1.1.4, General Design Criterion 4 – Environmental and Missile Design Bases. As described in this FSAR section, CPNPP conformance to the requirements of GDC-4 is described in the following:

  - Environmental Design of Mechanical and Electrical Equipment (FSAR Section 3.11)

  - Protection Against the Dynamic Effects Associated with the Postulated Rupture of Piping (FSAR Section 3.6)

  - Missile Protection (FSAR Section 3.5)

- GDC-17 is addressed in FSAR Section 3.1.2.8 for electrical power supply for equipment important to safety. GDC-17 requires that the plant safety-related systems and components be provided with offsite and onsite power to accommodate operating conditions associated with normal operation, maintenance, testing, and postulated accidents.

- GDC-60 is addressed in FSAR Section 3.1.6.1. GDC-60 requires that the plant design include means to suitably control the release of radioactive materials in gaseous effluents and that sufficient holdup capacity be provided for retention of gaseous effluents containing radioactive materials. FSAR Section 3.1.6.1 states that the handling, control, and release of radioactive materials during Modes 1 and 2 is in compliance with 10 CFR 50, Appendix I, and is described in the Offsite Dose Calculation Manual.

2.7.6.2 Technical Evaluation

2.7.6.2.1 Introduction

The ESF ventilation systems function to maintain temperatures within specified limits in areas containing safety-related equipment. Normal ventilation exhausts from potentially contaminated areas are filtered and the discharge is monitored for radiation. Included in the scope of the essential ventilation system are the following subsystems:

- FSAR Section 9.4.5, Engineered Safety Features (Safeguards) Building Ventilation
- FSAR Section 9.4.3, Auxiliary Building Ventilation

- FSAR Section 9.4.2, Fuel Building Ventilation
- FSAR Section 9.4C.1, Diesel Generator Building Ventilation

The engineered safety feature (ESF, safeguards) ventilation system has a non-safety supply heating, ventilation, and air conditioning (HVAC) subsystem (primary plant ventilation) that provides clean, filtered, and tempered air to the areas of the buildings. The system exhausts air from the equipment rooms and open areas of the buildings through a closed exhaust system. The exhaust system includes modular 100-percent capacity banks of high-efficiency particulate air (HEPA) filters, iodine adsorbers, and redundant 100-percent capacity fans discharging to the atmosphere. Exhaust air from ESF areas is monitored by in-line radiation monitors. This arrangement ensures the proper direction of airflow for removal of airborne radioactivity from the safety feature areas. In addition to the main safety feature ventilation system, the various ESF pumps and electrical areas are provided with additional emergency cooling coil and fan unit feeds from the plant safety chilled water system when the pumps are operating.

The spent fuel pool area ventilation system is a part of the Fuel Building ventilation system. Refer to Licensing Report (LR) subsection 2.7.4, Spent Fuel Pool Area Ventilation System, for the evaluation of this system.

The Auxiliary Building and radwaste area ventilation system is evaluated in LR subsection 2.7.5, Auxiliary and Radwaste Area and Turbine Area Ventilation System.

The diesel generators are housed in adjacent, but separate rooms, each of which is serviced by a safety-related ventilation system having inlet and exhaust wall louvers, and each room has a set of four 25-percent capacity exhaust fans drawing outside air across the room.

2.7.6.2.2 Description of Analyses and Evaluations

The changes in heat loads for ventilation subsystems in areas served by the ESF ventilation system were evaluated to ensure that the ventilation systems are capable of performing their intended functions under SPU conditions. The changes were found to be insignificant to degrade essential system operation, to impact the system's capability to circulate sufficient air to prevent accumulation of flammable or explosive gases, or to impact its ability to control airborne particulate material accumulation.

Other evaluations related to the ESF ventilation systems are addressed in the following LR sections:

- Protection against dynamic effects, including GDC-4 requirements, of missiles, pipe whip and discharging fluids – LR subsection 2.2.1, Pipe Rupture Locations and Associated Dynamic Effects; and LR subsection 2.5.1.3, Pipe Failures.

- Electrical equipment qualification – LR subsection 2.3.1, Environmental Qualification of Electrical Equipment.

- Onsite and offsite electric power systems, including GDC-17 requirements – LR subsection 2.3.2, Offsite Power System; and LR Section 2.3.3, AC Onsite Power System.

- Protection against turbine missiles and internal missiles is discussed in LR subsection 2.5.1.2, Missile Protection.

- The Control Room ventilation system, including the relay rooms, and the Technical Support Center ventilation system are evaluated in LR subsection 2.7.1, Control Room Habitability System.

2.7.6.2.3 Results

The ESF Building area temperature does not increase after implementation of the SPU. The insignificant increase in heat load in these buildings is primarily due to the changes in the piping systems operating conditions.

The auxiliary and radwaste area temperatures do not increase after implementation of the SPU. The increase heat load in the Auxiliary Building areas is primarily due to the changes in the main steam and feedwater system operating temperatures. The Auxiliary Building clean side uses outside air exchange to provide cooling; outside air temperature changes dominate any potential temperature semi-changes caused by the SPU.

The diesel generator loading is not increased after implementation of the SPU (Refer to LR subsection 2.3.3, AC Onsite Power System). Therefore, the ventilation system's ability to provide the required temperature conditions for personnel and equipment is not impacted by the SPU.

The evaluation of the plant equipment changes for the proposed SPU did not identify any need to modify the ESF essential ventilation systems. There are no equipment changes as a result of the SPU that could create a new potentially unmonitored radioactive release path. Thus, following the SPU, CPNPP will continue to meet the current licensing basis with respect to GDC-60. The effects of potential releases to the environment are evaluated in LR subsection 2.10.1, Occupational and Public Radiation Doses, and remain within current limits following the SPU.

SPU activities neither add any new components, nor do they introduce any new functions for existing components that would change the licensed system evaluation boundaries. The SPU does not add any new or previously unevaluated materials to the system because no modifications are necessary for the ESF ventilation system components. System component internal and external environments remain within the parameters previously evaluated.

2.7.6.3 Conclusions

The Luminant Power review of the ESF (safeguards) ventilation system has adequately accounted for the effects of the proposed SPU on the ability of the ESF (safeguards) ventilation system to provide a suitable and controlled environment for ESF components. The Luminant Power review further concluded that the ESF (safeguards) ventilation system will continue to control the release of gaseous radioactive effluents to the environment following implementation of the proposed SPU. Based on this, Luminant Power concludes that the ESF (safeguards) ventilation system will continue to meet the CPNPP current licensing basis with respect to the requirements of GDCs-4, -17, and -60. Therefore, Luminant Power finds the proposed SPU acceptable.

2.7.7 Other Ventilation Systems (Containment)

2.7.7.1 Regulatory Evaluation

The functions of the containment ventilation system are to provide heat removal from the containment atmosphere, to remove radioactive materials from the containment atmosphere, and to provide containment pressure control under normal and accident conditions. The Luminant Power review of the containment structure ventilation system focused on the effects that the proposed stretch power uprate (SPU) will have on the functional performance of the system. The acceptance criteria for the containment structure ventilation system are based on:

- General Design Criterion (GDC)-2, insofar as it requires that safety-related structures, systems, and components (SSCs) be designed to accommodate the effects of withstanding the effects of earthquakes.
- GDC-4, insofar as it requires that safety-related SSCs be designed to accommodate the effects of, and be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.
- GDC-17, insofar as it requires onsite and offsite electric power systems be provided to permit functioning of safety-related SSCs.
- GDC-19, insofar as it requires that equipment at appropriate locations outside the Control Room be provided with: (1) the capability for prompt hot shutdown of the reactor, and (2) a potential capability for subsequent cold shutdown of the reactor.
- GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents.
- GDC-61, insofar as it requires that systems containing radioactivity be designed with appropriate confinement and containment.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

The adequacy of the CPNPP containment ventilation design is assessed relative to conformance with the following:

- GDC-4 is described in FSAR Section 3.1.1.4, GDC-4 – Environmental and Missile Design Bases. As described in this FSAR Section, CPNPP received post-construction review as part of the Systematic Evaluation Program (SEP). Conformance to the requirements of GDC-4 is described in the following:

- Environmental Design of Mechanical and Electrical Equipment (FSAR Section 3.11)

- Protection Against the Dynamic Effects Associated with the Postulated Rupture of Piping (FSAR Section 3.6)

- Missile Protection (FSAR Section 3.5)

- GDC-17 is addressed in FSAR Section 3.1.2.8 for electrical power supply for equipment important to safety. GDC-17 requires that the plant safety-related systems and components be provided with offsite and onsite power to accommodate operating conditions associated with normal operation, maintenance, testing, and postulated accidents.

- GDC-60 is addressed in FSAR Section 3.1.6.1. GDC-60 requires that the plant design include means to suitably control the release of radioactive materials in gaseous effluents and that sufficient holdup capacity be provided for retention of gaseous effluents containing radioactive materials. FSAR Section 3.1.6.1 states that the handling, control, and release of radioactive materials during Modes 1 and 2 is in compliance with 10 CFR 50, Appendix I, and is described in the Offsite Dose Calculation Manual.

- GDC-61 is addressed in FSAR Section 3.1.6.2. The requirements of GDC-61 relevant to the containment ventilation system and gaseous effluents treatment system require that the ventilation system be designed to assure adequate safety under normal and postulated accident conditions. The system shall be designed (1) to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, and (3) with appropriate containment, confinement, and filtering systems.

  FSAR Section 3.1.6.2 states that the gaseous waste management system is designed to ensure adequate safety under normal and postulated accident conditions by providing the following:

  - Components are designed and located such that appropriate periodic inspection and testing may be performed

  - All areas of the plant are designed with suitable shielding for radiation protection based on anticipated radiation dose rates and occupancy as discussed in FSAR Chapter 12

2.7.7.2 Technical Evaluation

2.7.7.2.1 Introduction

The containment ventilation systems are described in FSAR Section 9.4A.1. The containment ventilation system is designed to accomplish the following:

- Remove the normal heat loss from the equipment and piping in the containment during plant operation and maintain a normal ambient temperature at or below 120°F.
- Provide sufficient air circulation and filtering throughout all containment areas to permit safe and continuous access to the reactor containment within specified hours after reactor shutdown.
- Provide for positive circulation of air across the refueling water surface to ensure personnel access and safety during shutdowns.
- Provide a minimum containment ambient temperature of 50°F during reactor shutdown.
- Provide for purging of the containment to the plant ventilation discharge vent to atmosphere for dispersion to the environment as allowed by applicable regulations.

Included within the scope of the containment system are the following subsystems:

- *Containment air recirculation and cooling system*
- Control rod drive mechanism cooling system
- Neutron detector well cooling system
- Containment preaccess filtration system
- Containment purge supply and exhaust system
- Containment pressure relief system
- Reactor coolant pipe penetration cooling subsystem

The principal components of the containment ventilation system include filters, fans, dampers, valves, heat exchangers, and essential ductwork and piping.

The containment recirculation fans, control rod drive mechanism fans, and reactor coolant pipe penetration cooling fans are direct-driven units, each set with standby units for redundancy. Each fan in the associated systems is provided with flow-indicating switches to verify existence of airflow. The containment recirculation fan cooler electrical connections and other equipment in the containment necessary for operation of the system are capable of operating under the environmental conditions following a loss-of-coolant accident (LOCA).

The control rod drive mechanism cooling system consists of fans, cooling coils, dampers, and ductwork that draw air through the control rod drive mechanism shroud and ejects it to the main containment volume. The reactor coolant pipe penetration system consists of a plenum, fans, outlet isolation dampers, and ductwork arranged to supply cool air to the hot and cold legs pipe

penetration supports. The reactor coolant pipe penetration cooling system is designed to prevent the bulk pipe tunnel concrete temperature from exceeding 200°F. The neutron detector well cooling system is a closed loop system and consists of cooling units with fans, cooling coils, isolation dampers, and provides cooled air to the annulus between the reactor vessel and the primary shield, neutron detectors, and to the nozzle supports. The containment preaccess filtration system's purpose is to absorb radioactive iodine vapor and radioactive particles that may occur as a result of normal primary system leakage inside the containment. The containment preaccess filtration system unit consists of a fan, a roughing filter, high-energy-efficiency air (HEPA) filters, and one iodine adsorber, with ductwork connected to the suction side of the unit and the discharge air is directed at the suction side of the containment recirculation system, which supplies the air through ductwork and guarantees proper mixing. The containment purge supply and exhaust system is a part of the primary plant ventilation systems modular arrangement. The supply subsystem includes an outside air connection to roughing filters, heating/cooling coils, and fans. The exhaust subsystem includes a prefilter, HEPA filters, and an iodine adsorber and exhaust fan, and releases air to the atmosphere via the plant ventilation discharge vent.

2.7.7.2.2 Description of Analyses and Evaluations

The changes in heat loads for ventilation subsystems in the containment were evaluated to ensure that the ventilation systems are capable of performing their intended functions under normal SPU modes.

Other evaluations related to the containment ventilation system are addressed in the following Licensing Report (LR) sections:

- Protection against dynamic effects, including GDC-4 requirements, of missiles, pipe whip, and discharging fluids – LR subsection 2.2.1, Pipe Rupture Locations and Associated Dynamic Effects; and LR subsection 2.5.1.3, Pipe Failures.
- Evaluation of the control rod drive mechanism cooling system – LR subsection 2.2.2.4, Control Rod Drive Mechanism.
- Electrical equipment qualification – LR subsection 2.3.1, Environmental Qualification of Electrical Equipment.
- Onsite and offsite electric power systems, including GDC-17 requirements – LR subsection 2.3.2, Offsite Power System and LR Section 2.3.3, AC Onsite Power System.
- Protection against turbine missiles and internal missiles is discussed in LR subsection 2.5.1.2, Missile Protection.
- Containment post-accident heat removal – LR subsection 2.6.1, Primary Containment Functional Design.

- Radiological consequences analysis – LR subsection 2.9.7, Radiological Consequences of a Design Basis Loss-of-Coolant Accident.
- Impact of containment purge related to normal operational radwaste effluents and associated doses – LR subsection 2.10.1, Occupational and Public Radiation Doses.

2.7.7.2.3 Results

The containment ventilation system's ability to provide the required temperature conditions for personnel and equipment in the containment during normal operation was evaluated. The results of the evaluation determined that an increase in the containment bulk air temperature of less than 0.15°F from current observed level will occur at SPU conditions. However, this increase in the normal operating containment bulk air temperature will not exceed the maximum normal operating bulk temperature limit of 120°F.

As a result of the SPU, there will be only minor temperature changes in the process fluids contained in these systems. The minor increase in heat loads can be adequately compensated for by the existing cooling coil units. Thus, no changes are required for the containment cooling system as a result of the SPU. The ability of the containment ventilation system to provide sufficient air circulation and cooling for access is not impacted by the SPU.

Related to post-accident operation, refer to LR subsection 2.6.1, Primary Containment Functional Design, for the system evaluation following an accident.

SPU activities neither add any new components nor do they introduce any new functions for existing components that would change the licensed system evaluation boundaries. Operating the containment ventilation system at SPU conditions does not add any new or previously unevaluated materials to the system. System component internal and external environments remain within the parameters previously evaluated.

2.7.7.3 Conclusions

The Luminant Power review of the containment ventilation system has adequately accounted for the effects of the proposed SPU on the ability of the containment ventilation system to provide a suitable and controlled environment for the containment components. Based on this, Luminant Power concludes that the containment ventilation system will continue to meet the CPNPP current licensing basis with respect to the requirements of GDCs-2, -4, -17, -19, -60 and -61. Therefore, Luminant Power finds the proposed SPU acceptable with respect to the containment ventilation system.