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SAFETY ANALYSIS REPORT

FOR

VERMONT YANKEE NUCLEAR POWER STATION

CONSTANT PRESSURE POWER UPRATE

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Glossary of Terms

Term	Definition
AC	Alternating Current
ACS	Alternate Cooling System
ADS	Automatic Depressurization System
AL	Analytical Limit
ALARA	As Low As Reasonably Achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence (Moderate Frequency Transient Event)
AOP	Abnormal Operating Procedure
APRM	Average Power Range Monitor
ARI	Alternate Rod Insertion
ARTS	Average Power Range Monitor / Rod Block Monitor / Technical Specifications
ASME	American Society of Mechanical Engineers
AST	Alternative Source Term
ATWS	Anticipated Transient Without Scram
BHP	Brake Horsepower
BIIT	Boron Injection Initiation Temperature
BOP	Balance-of-Plant
BPWS	Banked Position withdrawal Sequence
BWR	Boiling Water Reactor
BWROG	BWR Owners Group
BWRVIP	BWR Vessel and Internals Project
CAD	Containment Atmosphere Dilution
CDF	Core Damage Frequency
CFD	Condensate Filter Demineralizer
CFR	Code of Federal Regulations
CHUG	CHECWORKS™ Users Group
CLTP	Current Licensed Thermal Power

Term	Definition
CLTR	Constant Pressure Power Uprate Licensing Topical Report
CO	Condensation Oscillation
COLR	Core Operating Limits Report
CPPU	Constant Pressure Power Uprate
CRD	Control Rod Drive
CRDA	Control Rod Drop Accident
CREVS	Control Room Emergency Ventilation System
CRHZ	Control Room Habitability Zone
CSC	Containment Spray Cooling
CS	Core Spray
CST	Condensate Storage Tank
CT	Current Transformer
CUF	Cumulative Usage Factor
DBA	Design Basis Accident
DC	Direct Current
DHR	Decay Heat Removal
DW	Drywell
DWT	Deadweight
ECCS	Emergency Core Cooling System
EFPY	Effective Full Power Year
EOP	Emergency Operating Procedure
EPRI	Electric Power Research Institute
EPU	Extended Power Uprate
EQ	Environmental Qualification
ERFIS	Emergency Response Facility Information System
ES	Extraction Steam
FAC	Flow-Accelerated Corrosion
FFWTR	Final Feedwater Temperature Reduction
FHA	Fuel Handling Accident

Term	Definition
FIV	Flow Induced Vibration
FIVE	Fire Induced Vulnerability Evaluation
FOA	Forced Oil and Air
FOST	Fuel Oil Storage Tank
FPC	Fuel Pool Cooling
FPCC	Fuel Pool Cooling and Cleanup
FPCDS	Fuel Pool Cooling and Demineralizer System
FW	Feedwater
FWLC	Feedwater Level Control
GE	General Electric Company
GENE	General Electric Nuclear Energy
GE SIL	General Electric Service Information Letter
GL	Generic Letter
GNF	Global Nuclear Fuel - LLC
HCTL	Heat Capacity Temperature Limit
HCLPF	High Confidence of Low Probability of Failure
HELB	High Energy Line Break
HEM	Homogeneous Equilibrium Model
HEP	Human Error Probability
HEPA	High Efficiency Particulate Air
Hg_a	Inches of Mercury Absolute
HP	High Pressure
HPCI	High Pressure Coolant Injection
HPCS	High Pressure Core Spray
HPT	High-Pressure Turbine
HRA	Human Reliability Analysis
HVAC	Heating, Ventilation, and Air Conditioning
IASCC	Irradiation-Assisted Stress Corrosion Cracking

Term	Definition
ICF	Increased Core Flow
IEB	Inspection and Enforcement Bulletin
IEEE	Institute of Electrical and Electronics Engineers
IGSCC	Intergranular Stress Corrosion Cracking
ILBA	Instrument Line Break Accident
IORV	Inadvertent Opening of a Relief Valve
IPB	Isolated Phase Bus
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination of External Events
IRM	Intermediate Range Monitor
LAR	License Amendment Request
LBPCT	Licensing Basis Peak Cladding Temperature
LCS	Leakage Control System
LDS	Leak Detection System
LERF	Large Early Release Frequency
LHGR	Linear Heat Generation Rate
LOCA	Loss-of-Coolant Accident
LOFW	Loss of Feedwater Flow
LOOP	Loss of Offsite Power
LPCI	Low Pressure Coolant Injection
LPCS	Low Pressure Core Spray
LPRM	Local Power Range Monitor
LPSP	Low Power Setpoint
LPZ	Low Population Zone
LTP	Long Term Program
LTR	Licensing Topical Report
MAPLHGR	Maximum Average Planar Linear Heat Generation Rate
MCPR	Minimum Critical Power Ratio

Term	Definition
MELB	Moderate Energy Line Break
MELLLA	Maximum Extended Load Line Limit Analysis
MeV	Million Electron Volts
Mlb	Millions of Pounds
MOV	Motor Operated Valve
MS	Main Steam
MSIV	Main Steam Isolation Valve
MSIVC	Main Steam Isolation Valve Closure
MSIVD	Main Steam Isolation Valve Closure with Direct Scram event
MSIVF	Main Steam Isolation Valve Closure with Scram on High Flux
MSL	Main Steam Line
MSLBA	Main Steam Line Break Accident
MSVV	Main Steam Valve Vault
MVA	Million Volt Amps
MWe	Megawatts-Electric
MWt	Megawatt-Thermal
N/A	Not Applicable
NDE	Nondestructive Examination
NEI	Nuclear Energy Institute
NFPCS	Normal Fuel Pool Cooling Subsystem
NPDES	National Pollutant Discharge Elimination System
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NTSP	Nominal Trip Setpoint
NUREG	Nuclear Regulations
OFS	Orificed Fuel Support
OLTP	Original Licensed Thermal Power

Term	Definition
ON	Off Normal
OOS	Out-of-Service
OT	Operational Transient
ΔP	Differential Pressure –(psi)
P	Percent of CLTP
P_{25}	25% of CPPU Rated Thermal Power
PCAC	Primary Containment and Atmospheric Control
PCPL	Pressure Suppression Pressure Limit
PCS	Pressure Control System
PCT	Peak Cladding Temperature
PLHGR	Planar Linear Heat Generation Rate
PRA	Probabilistic Risk Assessment
PRFO	Pressure Regulator Failure – Open
PSA	Probabilistic Safety Analysis
psi	Pounds per square inch
psia	Pounds per square inch - absolute
psid	Pounds per square inch - differential
psig	Pounds per square inch - gauge
RBCCW	Reactor Building Closed Cooling Water
RBM	Rod Block Monitor
RCIC	Reactor Core Isolation Cooling
RCIS	Rod Control and Information System
RCPB	Reactor Coolant Pressure Boundary
RFP	Reactor Feedwater Pump
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
RIPD	Reactor Internal Pressure Difference(s)
RPT	Recirculation Pump Trip

Term	Definition
RPV	Reactor Pressure Vessel
RLB	Recirculation Line Break
RMSS	Reactor Mode Selector Switch
RRS	Reactor Recirculation System
RSIC	Radiation Shielding Information Center
RTP	Rated Thermal Power
RT_{NDT}	Reference Temperature of Nil-Ductility Transition
RWCU	Reactor Water Cleanup
RWM	Rod Worth Minimizer
S_m	Code Allowable Stress Limit
SAMG	Severe Accident Management Guideline
SAR	Safety Analysis Report
SBO	Station Blackout
SCM	Steam Condensing Mode
SDC	Shutdown Cooling
SER	Safety Evaluation Report
SHB	Shroud Bolt Head
SFP	Spent Fuel Pool
SFPCS	Standby Fuel Pool Cooling Subsystem
SGTS	Standby Gas Treatment System
SJAE	Steam Jet Air Ejectors
SLCS	Standby Liquid Control System
SLMCPR	Safety Limit Minimum Critical Power Ratio
SLO	Single-Loop Operation
SMA	Seismic Margins Assessments
SORV	Stuck-Open Relief Valve
SPC	Suppression Pool Cooling
SPDS	Safety Parameter Display System

Term	Definition
SRLR	Supplemental Reload Licensing Report
SRM	Source Range Monitor
SRP	Standard Review Plan
SRV	Safety Relief Valve
SRVDL	Safety Relief Valve Discharge Line
SSC	Systems, Structures and Components
SSV	Spring Safety Valve
TAF	Top of Active Fuel
TEDE	Total Effective Dose Equivalent
T-G	Turbine-Generator
TH	Thermal
TIP	Traversing Incore Probe
TLO	Two-Loop Operation
TSL	Technical Specification Instrument Limit
TSV	Turbine Stop Valve
TT	Turbine Trip
T_w	Time Available
UFSAR	Updated Final Safety Analysis Report
UHS	Ultimate Heat Sink
USE	Upper Shelf Energy
UT	Ultrasonic Testing
VYNPS	Vermont Yankee Nuclear Power Station
W	Percent of Recirculation Drive Flow
W_c	Percent of Rated Core Flow

EXECUTIVE SUMMARY

This report summarizes the results of all significant safety evaluations performed that justify uprating the licensed thermal power at the Vermont Yankee Nuclear Power Station (VYNPS). The requested license power level is an increase to 1912 MWt from the Current Licensed Thermal Power (CLTP) of 1593 MWt. This is the first power uprate for VYNPS.

GE has previously developed and implemented Extended Power Uprate (EPU). Based on the EPU experience, GE has developed an approach to uprating reactor power that maintains the current plant maximum normal operating reactor dome pressure. This approach is referred to as Constant Pressure Power Uprate (CPPU) and is contained in the Licensing Topical Report (LTR) NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," hereafter referred to as CLTR. The NRC approved the CLTR in the staff Safety Evaluation Report (SER) contained in the letter, William H. Ruland (NRC) to James F. Klapproth (GE), "Review of GE Nuclear Energy Licensing Topical Report, NEDC-33004P, Revision 3, 'Constant Pressure Power Uprate' (TAC No. MB2510)," dated March 31, 2003, for Boiling Water Reactor (BWR) plants containing GE fuel types and using GE accident analysis methods. VYNPS contains only GE fuel types and this evaluation uses only GE accident analysis methods. By performing the power uprate in accordance with the CLTR and within the constraints of the NRC SER, the evaluation of the plant safety analyses and system performance is reduced, thus allowing for a more streamlined process.

This report provides systematic application of the CLTR approach to VYNPS, including performance of plant-specific engineering assessments and confirmation of the applicability of the CLTR generic assessments required to support a CPPU.

It is not the intent of this report to explicitly address all the details of the analyses and evaluations described herein. For example, only previously NRC-approved or industry-accepted methods were used for the analyses of accidents and transients, as referred to in the CLTR. Therefore, the safety analysis methods have been previously addressed, and thus, are not explicitly addressed in this report. Also, event and analysis descriptions that are already provided in other licensing reports or the Updated Final Safety Analysis Report (UFSAR) are not repeated within this report. This report summarizes the significant evaluations needed to support a licensing amendment to allow for uprated power operation.

Uprating the power level of nuclear power plants can be done safely within plant-specific limits and is a cost-effective way to increase installed electrical generating capacity. Many light water reactors have already been uprated worldwide, including many BWR plants.

An increase in the electrical output of a BWR plant is accomplished primarily by generating and supplying higher steam flow to the turbine-generator. VYNPS, as originally licensed, has an as-designed equipment and system capability to accommodate steam flow rates above the current rating. Also, the plant has sufficient design margins to allow the plant to be safely uprated significantly beyond its originally licensed power level.

A higher steam flow is achieved by increasing the reactor power along specified control rod and core flow lines. A limited number of operating parameters are changed, some setpoints are adjusted and instruments are recalibrated. Plant procedures are revised, and tests similar to some of the original startup tests are performed.

Detailed evaluations of the reactor, engineered safety features, power conversion, emergency power, support systems, environmental issues, and design basis accidents were performed. This report demonstrates that VYNPS can safely operate at the requested CPPU level. However, non-safety power generation modifications must be implemented in order to obtain the electrical power output associated with the uprate power. Until these modifications are completed, the non-safety balance of plant equipment may limit the electrical power output, which in turn may limit the operating thermal power level to less than the Rated Thermal Power (RTP) level. These modifications have been evaluated and they do not constitute a material alteration to the plant, as discussed in 10 CFR 50.92.

The evaluations and reviews were conducted in accordance with the CLTR. The results of these evaluations and reviews are presented in the succeeding sections of this report:

- All safety aspects of VYNPS that are affected by the increase in thermal power were evaluated;
- Evaluations were performed using NRC-approved or industry-accepted analysis methods;
- No changes, which require compliance with more recent industry codes and standards, are being requested;
- The UFSAR will be updated for the CPPU related changes, after CPPU is implemented, per the requirements in 10 CFR 50.71(e);
- Limited hardware modifications (e.g., Reactor Core Isolation Cooling (RCIC) pipe supports, Residual Heat Removal (RHR) / Core Spray (CS) pump seal replacements, main steam line flow instruments, RHR Service Water motor cooling pipe rerouting) may be required to meet safety requirements, and any modification to power generation equipment will be implemented per 10 CFR 50.59;
- Systems and components affected by CPPU were reviewed to ensure there is no significant challenge to any safety system;
- Compliance with current VYNPS environmental regulations were reviewed;
- Potentially affected commitments to the NRC have been reviewed; and
- Planned changes not yet implemented have also been reviewed for the effects of CPPU.

1. INTRODUCTION

1.1 REPORT APPROACH

This report summarizes the results of all significant safety evaluations performed that justify uprating the licensed thermal power at the Vermont Yankee Nuclear Power Station (VYNPS). The requested license power level is an increase to 1912 MWt from the Current Licensed Thermal Power (CLTP) of 1593 MWt. This is the first power uprate for VYNPS.

GE has previously developed and implemented Extended Power Uprate (EPU). Based on the EPU experience, GE has developed an approach to uprating reactor power that maintains the current plant maximum normal operating reactor dome pressure. This approach is referred to as Constant Pressure Power Uprate (CPPU) and is contained in the Licensing Topical Report (LTR) NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," (Reference 1) hereafter referred to as the "CLTR." The NRC approved the CLTR in the staff Safety Evaluation Report (SER) contained in Reference 2 for Boiling Water Reactor (BWR) plants containing GE fuel types and using GE accident analysis methods. VYNPS contains only GE fuel types and this evaluation uses only GE accident analysis methods. By performing the power uprate in accordance with the CLTR and within the constraints of the NRC SER, the evaluation of the plant safety analyses and system performance is reduced, thus allowing for a more streamlined process.

This evaluation justifies a CPPU to 1912 MWt, which corresponds to 120% of CLTP for VYNPS. This report follows the generic format and content for CPPU licensing reports, as described in the CLTR.

1.1.1 Generic Assessments

Many of the component, system, and performance evaluations contained within this report have been generically evaluated in the CLTR, and found to be acceptable. The plant-specific applicability of these generic assessments is identified and confirmed in the applicable sections of this report. Generic assessments are those safety evaluations that can be dispositioned for a group or all BWR plants by:

- A bounding analysis for the limiting conditions,
- Demonstrating that there is a negligible effect due to CPPU, or
- Demonstrating that the required plant cycle specific reload analyses are sufficient and appropriate for establishing the CPPU licensing basis.

Bounding analyses may be based on either a demonstration that previous pressure increase power uprate assessments provided in Reference 3 or 4 (ELTR1 and ELTR2, respectively) are bounding or on specific generic studies provided in the CLTR. For these bounding analyses, the current CPPU experience is provided in the CLTR along with the basis and results of the assessment. For those CPPU assessments having a negligible effect, the current CPPU

experience plus a phenomenological discussion of the basis for the assessment is provided in the CLTR. For generic assessments that are fuel design dependent, the assessments are applicable to GE / Global Nuclear Fuel (GNF) fuel designs up through GE14, analyzed with GE methodology.

1.1.2 Plant-Specific Evaluation

Plant-specific evaluations are assessments of the principal evaluations that are not addressed by the generic assessments described in Section 1.1.1. The relative effect of CPPU on the plant-specific evaluations and the methods used for their performance are provided in this report. Where applicable, the assessment methodology is referenced. If a specific computer code is used, the name of this computer code is provided in the subsection. If the computer code is identified in Reference 1, 3, 4, or 5 these documents may be referenced rather than the original report. Table 1-1 provides a summary of the computer codes used.

1.2 PURPOSE AND APPROACH

An increase in electrical output of a BWR is accomplished primarily by generation and supply of higher steam flow to the turbine generator. Most BWRs, as originally licensed, have an as-designed equipment and system capability to accommodate steam flow rates at least 5% above the original rating. In addition, continuing improvements in the analytical techniques (computer codes) based on several decades of BWR safety technology, plant performance feedback, operating experience, and improved fuel and core designs have resulted in a significant increase in the design and operating margin between the calculated safety analyses results and the current plant licensing limits. The available margins in calculated results, combined with the as-designed excess equipment, system, and component capabilities (1) have allowed many BWRs to increase their thermal power ratings by 5% without any nuclear steam supply system (NSSS) hardware modification, and (2) provide for power increases up to 20% with some non-safety hardware modifications. These power increases involve no significant increase in the hazards presented by the plants as approved by the NRC at the original license stage.

The method for achieving higher power is to extend the power/flow map (Figure 1-1) along the Maximum Extended Load Line Limit Analysis (MELLLA). However, there is no increase in the maximum normal operating reactor vessel dome pressure or the maximum licensed core flow over their pre-CPPU values. CPPU operation does not involve increasing the maximum normal operating reactor vessel dome pressure, because the plant, after modifications to non-safety power generation equipment, has sufficient pressure control and turbine flow capabilities to control the inlet pressure conditions at the turbine.

1.2.1 Uprate Analysis Basis

VYNPS is currently licensed at the 100% of CLTP level of 1593 MWt. The CPPU RTP level included in this evaluation is 120% of the CLTP. Plant-specific CPPU parameters are listed in Table 1-2. The CPPU safety analyses are based on a power level of 1.02 times the CPPU power level unless the Regulatory Guide (RG) 1.49 two percent power factor is already accounted for

in the analysis methods consistent with the methodology described in Reference 5, or RG 1.49 does not apply (e.g., an Anticipated Transient Without Scram (ATWS) event).

1.2.2 Computer Codes

NRC-approved or industry-accepted computer codes and calculational techniques are used to demonstrate compliance with the applicable regulatory acceptance criteria. The application of these codes to the CPPU analyses complies with the limitations, restrictions, and conditions specified in the approving NRC SER where applicable for each code. The limitations on use of these codes and methods as defined in the NRC staff position letter reprinted in ELTR1 were followed for this CPPU analysis. Any exceptions to the use of the code or conditions of the applicable SER are noted in Table 1-1. The application of the computer codes in Table 1-1 is consistent with the current VYNPS licensing basis except where noted in this report.

1.2.3 Approach

The planned approach to achieving the higher power level consists of the change to the VYNPS licensing and design basis to increase the licensed power level to 1912 MWt, consistent with the approach outlined in the CLTR, except as specifically noted in this report. Consistent with the CLTR, the following plant-specific exclusions are exercised:

- No increase in maximum normal operating reactor dome pressure
- No increase in the maximum licensed core flow
- No increase to the currently requested (via Reference 6) MELLLA upper boundary

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The plant-specific evaluations are based on a review of plant design and operating data, as applicable, to confirm excess design capabilities; and, if necessary, identify required modifications associated with CPPU. All changes to the plant licensing basis have been identified in this report. For specified topics, generic analyses and evaluations in the CLTR demonstrate plant operability and safety. The dispositions in the CLTR are based on a 20% of CLTP increase, which is the requested power increase for VYNPS. For this increase in power, the conclusions of system/component acceptability stated in the CLTR are bounding and have been confirmed for VYNPS. The scope and depth of the evaluation results provided herein are established based on the

[1] The AST evaluations (Reference 28) were performed at the CPPU RTP. This is consistent with CLTR Section 1.0 and the associated NRC SER.

approach in the CLTR and unique features of the plant. The results of these evaluations are presented in the following sections:

(a) **Reactor Core and Fuel Performance:** Specific analyses required for CPPU have been performed for a representative fuel cycle with the reactor core operating at CPPU conditions. Specific core and fuel performance is evaluated for each operating cycle, and will continue to be evaluated and documented for the operating cycles that implement CPPU.

(b) **Reactor Coolant System and Connected Systems:** Evaluations of the NSSS components and systems have been performed at CPPU conditions. These evaluations confirm the acceptability of the effects of the higher power and the associated change in process variables (i.e., increased steam and feedwater flows). Safety-related equipment performance is the primary focus in this report, but key aspects of reactor operational capability are also included.

(c) **Engineered Safety Feature Systems:** The effects of CPPU power operation on the Containment, Emergency Core Cooling System (ECCS), Standby Gas Treatment system and other Engineered Safety Features have been evaluated for key events. The evaluations include the containment responses during limiting Anticipated Operational Occurrences (AOOs) and special events, ECCS - Loss-of-Coolant Accident (LOCA), and Safety Relief Valve (SRV) containment dynamic loads.

(d) **Control and Instrumentation:** The control and instrumentation signal ranges and analytical limits for setpoints have been evaluated to establish the effects of the changes in various process parameters such as power, neutron flux, steam flow and feedwater flow. As required, setpoint evaluations have been performed to determine the need for any Technical Specification setpoint changes for various functions (e.g., main steam line high flow isolation setpoints).

(e) **Electrical Power and Auxiliary Systems:** Evaluations have been performed to establish the operational capability of the plant electrical power and distribution systems and auxiliary systems to ensure that they are capable of supporting safe plant operation at the CPPU power level.

(f) **Power Conversion Systems:** Evaluations have been performed to establish the operational capability of various non-safety Balance-of-Plant (BOP) systems and components to ensure that they are capable of delivering the increased power output, and/or to identify the modifications necessary to obtain full CPPU power.

(g) **Radwaste Systems and Radiation Sources:** The liquid and gaseous waste management systems have been evaluated at limiting conditions for CPPU to show that applicable release limits continue to be met during operation at higher power. The radiological consequences have been evaluated for CPPU to show that applicable regulations have been met for the CPPU power conditions. This evaluation includes the effect of higher power level on source terms, on-site doses and off-site doses, during normal operation.

(h) **Reactor Safety Performance Evaluations:** The limiting Updated Final Safety Analysis Report (UFSAR) analyses for design basis events have been addressed as part of the CPPU evaluation. All limiting accidents, AOOs, and special events have been analyzed or generically dispositioned consistent with the CLTR and show continued compliance with regulatory requirements. [[

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(i) **Additional Aspects of CPPU:** High-Energy Line Break (HELB) and environmental qualification evaluations have been performed at bounding conditions for CPPU to show the continued operability of plant equipment under CPPU conditions. The effects of CPPU on the VYNPS Individual Plant Evaluation (IPE) have been analyzed to demonstrate that there are no new vulnerabilities to severe accidents.

1.3 CPPU PLANT OPERATING CONDITIONS

1.3.1 Reactor Heat Balance

The operating pressure, the total core flow, and the coolant thermodynamic state characterize the thermal hydraulic performance of a BWR reactor core. The CPPU values of these parameters are used to establish the steady state operating conditions and as initial and boundary conditions for the required safety analyses. The CPPU values for these parameters are determined by performing heat (energy) balance calculations for the reactor system at CPPU conditions.

The reactor heat balance relates the thermal-hydraulic parameters to the plant steam and feedwater flow conditions for the selected core thermal power level and operating pressure. Operational parameters from actual plant operation are considered (e.g., steam line pressure drop) when determining the expected CPPU conditions. The thermal-hydraulic parameters define the conditions for evaluating the operation of the plant at CPPU conditions. The thermal-hydraulic parameters obtained for the CPPU conditions also define the steady state operating conditions for equipment evaluations. Heat balances at appropriately selected conditions define the initial and boundary conditions for plant safety analyses.

Figure 1-2 shows the CPPU heat balance at 100% of CPPU RTP and 100% rated core flow. Figure 1-3 shows the CPPU heat balance at 102% of CPPU RTP and 100% rated core flow.

Table 1-2 provides a summary of the reactor thermal-hydraulic parameters for the current rated and CPPU conditions. At CPPU conditions, the maximum nominal operating reactor vessel dome pressure is maintained at the current value, which minimizes the need for plant and licensing changes. With the increased steam flow and associated non-safety BOP modifications, the current dome pressure provides sufficient operating turbine inlet pressure to assure good pressure control characteristics.

1.3.2 Reactor Performance Improvement Features

The UFSAR, core fuel reload evaluations, and the Technical Specifications currently include allowances for plant operation with the performance improvement features and the equipment Out-of-Service (OOS) listed in Table 1-2. When limiting, the input parameters related to the performance improvement features or the equipment OOS have been included in the safety analyses for CPPU. The use of these performance improvement features and allowing for equipment OOS is continued during CPPU operation. The evaluations that are dependent upon cycle length are performed for CPPU assuming an 18-month cycle.

1.4 SUMMARY AND CONCLUSIONS

This evaluation has covered a CPPU to 120% of CLTP. The strategy for achieving higher power is to extend the MELLLA power/flow map region along the upper boundary extension.

The VYNPS licensing requirements have been reviewed to demonstrate how this uprate can be accommodated without a significant increase in the probability or consequences of an accident previously evaluated, without creating the possibility of a new or different kind of accident from any accident previously evaluated, and without exceeding any existing regulatory limits or design allowable limits applicable to the plant which might cause a reduction in a margin of safety. The CPPU described herein involves no significant hazard consideration.

Table 1-1
Computer Codes Used For CPPU *

Task	Computer Code	Version or Revision	NRC Approved	Comments
Anticipated Transient Without Scram	ODYN STEMP PANACEA	10 04 10	Y (1) Y	NEDE-24154P-A Suppl. 1, Vol. 4 NEDE-30130-P-A
Appendix R Fire Protection	R5YAB1A FROSSTEY2 GOTHIC	1.0 29 7.0	Y Y Y	VYNPS Tech Spec. 6.6.C VYNPS Tech Spec. 6.6.C (2)
Containment System Response (3)	SHEX M3CPT LAMB	05 05 08	Y Y (4)	(5) NEDO-10320, Apr. 1971 NEDE-20566-P-A, Sept. 1986
ECCS-LOCA	LAMB GESTR SAFER ISCOR TASC	08 08 04 09 03A	Y Y Y Y (6) Y	NEDO-20566A NEDE-23785-1-PA, Rev. 1 (7) (8) (9) NEDE-24011P Rev. 0 SER NEDC-32084P-A, Rev. 2
Fission Product Inventory	ORIGEN2	2.1	N	Isotope Generation and Depletion Code
Nominal Reactor Heat Balance	ISCOR	09	Y (6)	NEDE-24011P Rev. 0 SER
Piping Evaluations	STEHAM HAMTOPC CHPLOT NUPIPE-SWPC PC-PREPS PILUG-PC	02 (L03) 01 (L00) 02 (L02) 01 (L00) 04 (L00) 00 (L00)	N (10) N (10) N (10) N (10) N (10) N (10)	STEHAM, HAMTOPC & CHPLOT are used to determine fluid transient forcing functions. NUPIPE-SWPC used to perform piping evaluations. PC-PREPS used to perform pipe support structural capability and baseplate assessments. PILUG-PC used to perform IWA analysis.
Reactor Core and Fuel Performance	TGBLA PANACEA ISCOR	04 10 09	Y Y Y (6)	NEDE-30130-P-A NEDE-30130-P-A NEDE-24011P Rev. 0 SER
Reactor Internal Pressure Differences	ISCOR LAMB TRACG	09 07 02	Y (6) (4) Y (11)	NEDE-24011P Rev. 0 SER NEDE-20566P-A NEDC-32176P, Rev. 2 NEDC-32177P, Rev. 2 NRC SER TAC No. M90270, Sept. 30, 1994
Reactor Recirculation System	BILBO	04V	(12)	NEDE-23504, Feb. 1977
Risk Assessment	MAAP RISKMAN	4.04 5.02	N N	(13) (13)

Task	Computer Code	Version or Revision	NRC Approved	Comments
RPV Fluence	TGBLA DORTG01	04 01	Y N	(14) (15), (16),(17)
Stability Analysis	PANACEA ISCOR TRACG ODYSY	10 09 02 05	Y Y (6) Y Y	NEDE-30130-P-A NEDE-24011P Rev. 0 SER NEDO-32465-A NEDE-32177P, Rev. 1, June 1993 NEDC-32992P-A
Transient Analysis	PANACEA ISCOR ODYN SAFER	10 09 10 04	Y Y (6) Y (18)	NEDE-30130P-A (19) NEDE-24011P, Rev. 0 SER NEDO-24154-A NEDC-32424P-A, NEDC-32523P-A, (7), (8), (9)

Notes:

* The application of these codes to the CPPU analyses complies with the limitations, restrictions, and conditions specified in the approving NRC SER where applicable for each code. The application of the codes also complies with the SERs for the EPU programs.

1. The STEMP code uses fundamental mass and energy conservation laws to calculate the suppression pool heatup. The use of STEMP was noted in NEDE-24222, "Assessment of BWR Mitigation of ATWS, Volume I & II (NUREG-0460 Alternate No. 3) December 1, 1979." The code has been used in ATWS applications since that time. There is no formal NRC review and approval of STEMP or the ATWS topical report.
2. GOTHIC 5.0e was accepted by the NRC for containment calculations in support of Amendment No. 163.
3. The methodology change from that used for the UFSAR analysis is discussed in Section 4.1.
4. The LAMB code is approved for use in ECCS-LOCA applications (NEDE-20566P-A and NEDO-20566A), but no approving SER exists for the use of LAMB in the evaluation of reactor internal pressure differences or containment system response. The use of LAMB for these applications is consistent with the model description of NEDE-20566P-A.
5. The application of the methodology in the SHEX code to the containment response is approved by NRC in the letter from A. Thadani (NRC) to G. L. Sozzi (GE), "Use of the SHEX Computer Program and ANSI/ANS 5.1-1979 Decay Heat Source Term for Containment Long-Term Pressure and Temperature Analysis," July 13, 1993.
6. The ISCOR code is not approved by name. However, the SER supporting approval of NEDE-24011P Rev. 0 by the May 12, 1978 letter from D. G. Eisenhut (NRC) to R. Gridley (GE) finds the models and methods acceptable, and mentions the use of a digital computer code. The referenced digital computer code is ISCOR. The use of ISCOR to provide core thermal-hydraulic information in reactor internal pressure differences, Transient, ATWS, Stability, and LOCA applications is consistent with the approved models and methods.

7. Letter, J.F. Klapproth (GE) to NRC, Transmittal of GE Proprietary Report NEDC-32950P, "Compilation of Improvements to GENE's SAFER ECCS-LOCA Evaluation Model," January 2000, by letter dated January 27, 2000.

8. Letter, S.A. Richards (NRC) to J.F. Klapproth (GE), "General Electric Nuclear Energy (GENE) Topical Reports GENE (NEDC)-32950P and GENE (NEDC)-32084P Acceptability Review," May 24, 2000.

9. "SAFER Model for Evaluation of Loss-of-Coolant Accidents for Jet Pump and Non-Jet Pump Plants," NEDE-30996P-A, General Electric Company, October 1987.

10. Stone and Webster proprietary computer codes STEHAM, HAMTOPC, CHPLOT, NUPIPE-SWPC, PILUG-PC, and PC-PREPS are not approved specifically by name by the NRC. These codes are qualified in accordance with Stone and Webster Quality Assurance Standard QS2.7 "Computer Software." These computer codes are qualified for use in QA Category I Nuclear Safety-Related applications.

11. NRC has reviewed and accepted the TRACG application for the flow-induced loads on the core shroud as stated in NRC SER TAC No. M90270.

12. Not a safety analysis code that requires NRC approval. The code application is reviewed and approved by GENE for "Level-2" application and is part of GENE's standard design process. Also, the application of this code has been used in previous power uprate submittals.

13. These code packages are standard industry-accepted codes for the development of Probabilistic Risk Assessment (PRA) models and calculations, which have been used to support NRC submittals for IPEs.

14. Letter, S.A. Richards (NRC) to G. A. Watford (GE), "Amendment 26 to GE Licensing Topical Report NEDE-24011-P-A, GESTAR II – Implementing Improved GE Steady-State Methods (TAC No. MA6481)," November 10, 1999.

15. NEDC-32983P-A "GE Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations," Revision 1, December 2001.

16. CCC-543, "TORT-DORT Two-and Three-Dimensional Discrete Ordinates Transport Version 2.8.14," Radiation Shielding Information Center (RSIC), January 1994.

17. Letter, S. A. Richards (NRC) to J. F. Klapproth (GE), "Safety Evaluation for NEDC-32983P, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation (TAC No. MA9891)," September 14, 2001.

18. The ECCS-LOCA codes are not explicitly approved for Transient usage. The staff concluded that SAFER is qualified as a code for best estimate modeling of loss-of-coolant accidents and loss of inventory events via the approval letter (Letter, C. O. Thomas (NRC) to J. F. Quirk (GE), "Review of NEDE-23785-1 (P), 'GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident, Volumes I and II'," August 29, 1983) and the evaluation for NEDE-23785P, Revision 1, Volume II. In addition, the use of SAFER in the analysis of long-term Loss-of-Feedwater events is specified in the approved LTRs for power uprate, i.e., ELTR1 and ELTR2.

19. The physics code PANACEA provides inputs to the transient code ODYN. The improvements to PANACEA that were documented in NEDE-30130-P-A were incorporated into ODYN by way of Amendment 11 of GESTAR II (NEDE-24011-P-A). The use of

PANAC Version 10 in this application was initiated following approval of Amendment 13 of GESTAR II by letter from G.C. Lainas (NRC) to J.S. Charnley (GE), MFN 028-086, "Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-A Amendment 13, Rev. 6 General Electric Standard Application for Reactor Fuel," March 26, 1998.

Table 1-2
CLTP and CPPU Plant Operating Conditions

Parameter	CLTP Value [1]	CPPU Value
Thermal Power (MWt)	1593	1912
Vessel Steam Flow [2] (Mlb/hr)	6.458	7.906
Full Power Core Flow Range		
Mlb/hr	36.0 to 51.4	47.5 to 51.4
% Rated	75.0 to 107.0	99.0 to 107.0
Maximum Normal Operating Dome Pressure (psia)	1025	No Change
Maximum Normal Operating Dome Temperature (°F)	547.9	No Change
Pressure at upstream side of Turbine Stop Valve (TSV) (psia)	983	961
Full Power Feedwater		
Flow (Mlb/hr)	6.430	7.878
Temperature (°F)	376.0	391.5
Core Inlet Enthalpy [3] (Btu/lb)	521.1	518.9

Notes:

1. Based on CLTP reactor heat balance.
2. At normal feedwater heating. VYNPS is not licensed for Final Feedwater Temperature Reduction and is not requesting this plant performance enhancement as part of CPPU.
3. At 100% core flow condition.
4. Currently licensed performance improvement features and/or equipment OOS that are included in CPPU evaluations:
   a. Average Power Range Monitor (APRM) / Rod Block Monitor (RBM) / Technical Specifications (ARTS) / MELLLA (License Amendment Request (LAR) submitted (Reference 6))
   b. Single-Loop Operation (SLO)
   c. One SRV OOS
   d. Increased Core Flow (ICF) of 107% rated core flow

Figure 1-1
Power/Flow Operating Map for CPPU

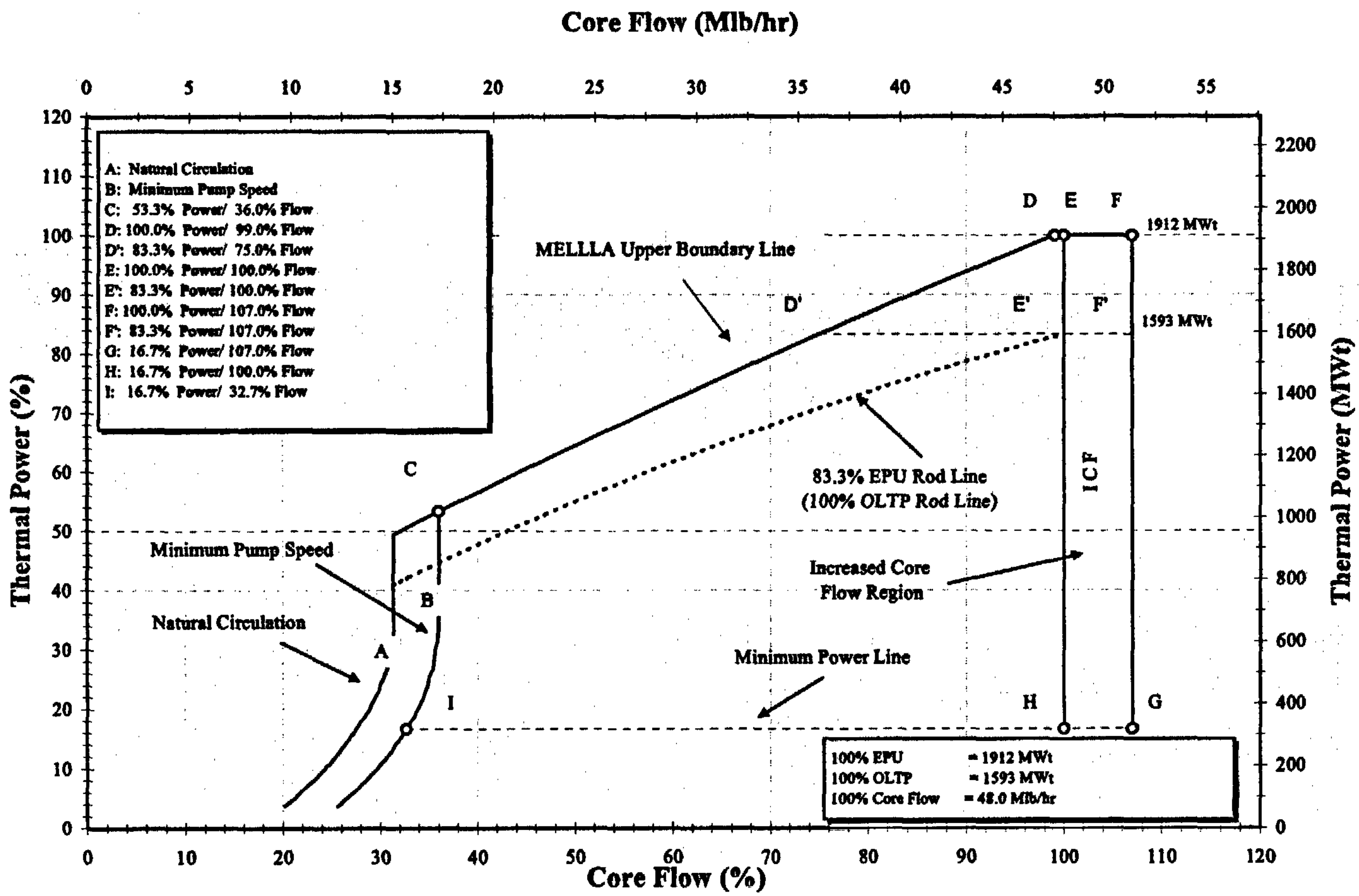
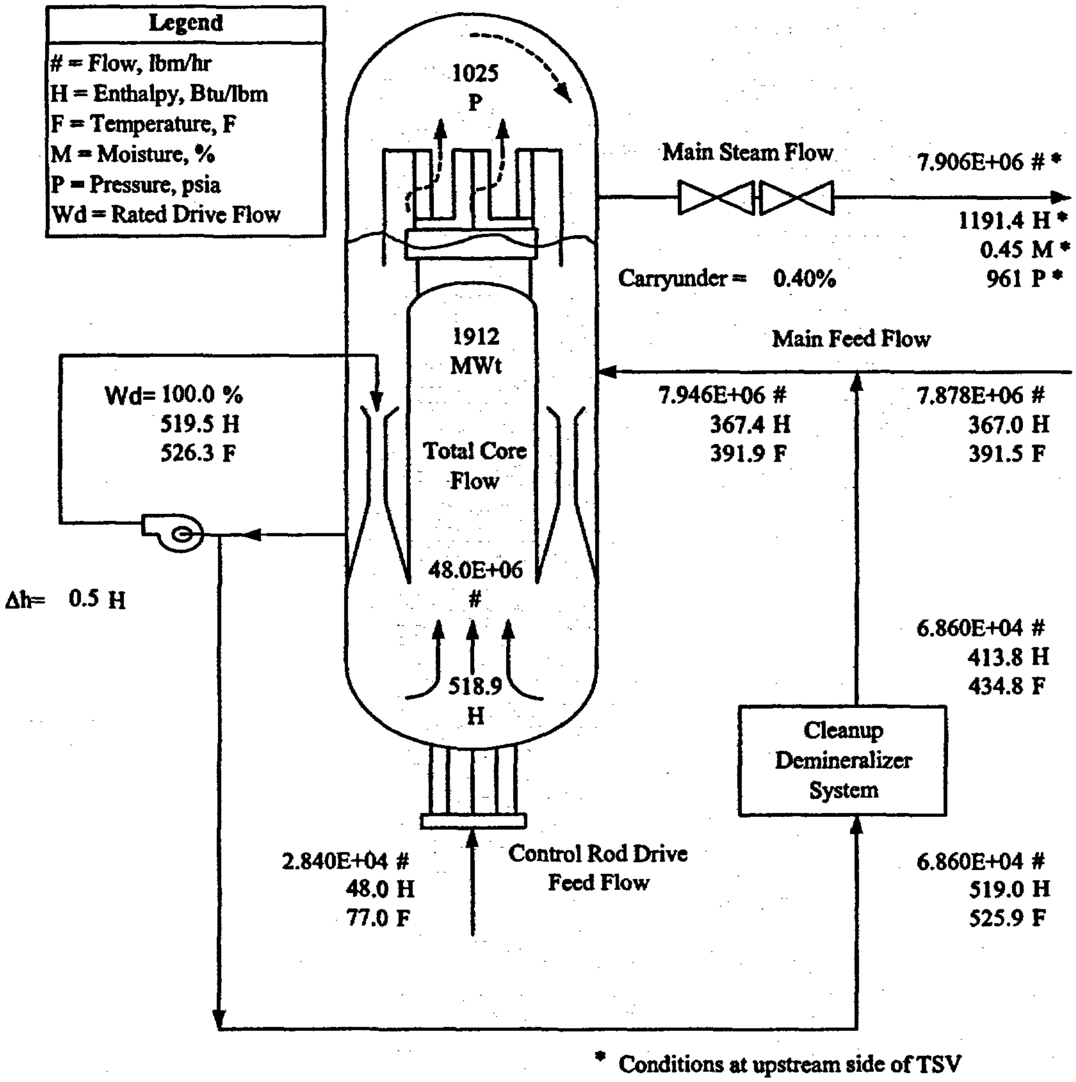
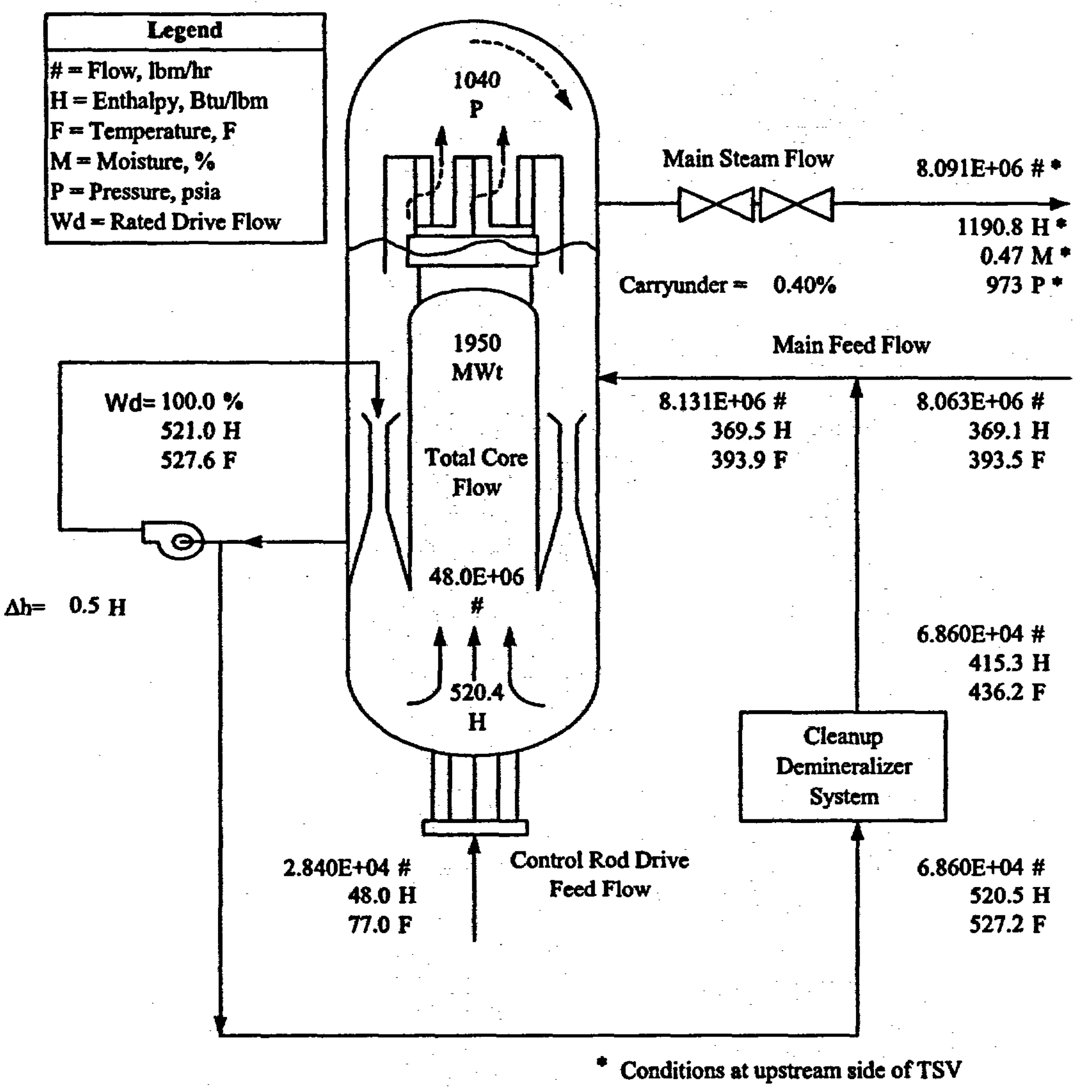


Figure 1-2
CPPU Heat Balance – Nominal
(@ 100% Power and 100% Core Flow)



Core Thermal Power	1912.0
Pump Heating	3.6
Cleanup Losses	-2.1
Other System Losses	-0.6
Turbine Cycle Use	1912.9 MWt

Figure 1-3
CPPU Heat Balance - Overpressure Protection Analysis
(@ 102% Power and 100% Core Flow)



Core Thermal Power	1950.0
Pump Heating	3.6
Cleanup Losses	-2.1
Other System Losses	-0.6
Turbine Cycle Use	1950.9 MWt

2. REACTOR CORE AND FUEL PERFORMANCE

This section primarily focuses on the information requested in RG 1.70, Chapter 4, applicable to CPPU. The reload process (Reference 5) will result in a plant-specific Supplemental Reload Licensing Report (SRLR) and Core Operating Limits Report (COLR).

2.1 FUEL DESIGN AND OPERATION

The effect of CPPU on the fuel design and core operation for VYNPS is described below. The topics addressed in this evaluation are:

Topic	CPPU Disposition	VYNPS Result
Fuel product line design	[[	
Core design		
Fuel thermal margin monitoring threshold		]]

CPPU increases the average power density proportional to the power increase and has some effects on operating flexibility, reactivity characteristics and energy requirements. The additional energy requirements for CPPU are met by an increase in bundle enrichment, an increase in the reload fuel batch size, and/or changes in fuel loading pattern to maintain the desired plant operating cycle length. The power distribution in the core is changed to achieve increased core power, while limiting the Minimum Critical Power Ratio (MCPR), Linear Heat Generation Rate (LHGR), and Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) in any individual fuel bundle to be within its operating limits as defined in the COLR.

The CLTP core at VYNPS consists only of GE fuel types. VYNPS transitioned to GE14 fuel in Cycle 23 and will continue to use only GE fuel types through CPPU implementation. [[

]] The fuel design limits are established for all new fuel product line designs as a part of the fuel introduction and reload analyses. [[

]]

The percent power level above which fuel thermal margin monitoring is required may change with CPPU. The original plant operating licenses set this monitoring threshold at a typical value of 25% of Rated Thermal Power (RTP). [[

]]

For CPPU, as specified in the CLTR, the fuel thermal margin monitoring threshold is scaled down, if necessary, to ensure that monitoring is initiated [[

]], then the existing power threshold value must be lowered by a factor of $1.2/P_{25}$.

For VYNPS, the CPPU fuel thermal monitoring threshold is established at 23% of CPPU RTP. A change in the fuel thermal monitoring threshold also requires a corresponding change to the Technical Specification reactor core safety limit for reduced pressure or low core flow.

2.2 THERMAL LIMITS ASSESSMENT

The effect of CPPU on the MCPR safety and operating limits and on the MAPLHGR and LHGR limits for VYNPS is addressed below. The topics addressed in this evaluation are:

Topic	CPPU Disposition	VYNPS Result
2.2.1 Safety Limit MCPR	[[	
2.2.2 MCPR Operating Limit		
2.2.3 MAPLHGR Limit		
2.2.3 Maximum LHGR Limit		]]

Operating limits ensure that regulatory and/or safety limits are not exceeded for a range of postulated events (e.g., transients, LOCA). This section addresses the effects of CPPU on thermal limits. A representative cycle core is used for the CPPU evaluation. Cycle-specific core configurations, evaluated for each reload, confirm CPPU capability, and establish or confirm cycle-specific limits, as is currently the practice.

2.2.1 Safety Limit MCPR

The Safety Limit MCPR (SLMCPR) can be affected slightly by CPPU due to the flatter power distribution inherent in the increased power level. [[

]] This effect is not changed by the CPPU approach (Reference 1). The SLMCPR analysis reflects the actual plant core-loading pattern and is performed for each plant reload core (Reference 5). [[

]]

2.2.2 MCPR Operating Limit

CPPU operating conditions [[ ]] The MCPR Operating Limit is calculated by adding the change in MCPR due to the limiting AOO event to the SLMCPR and is determined on a cycle specific basis. CPPU does not change the method used to determine this limit. The effect of CPPU on AOO events is addressed in Section 9.1. [[

]]

2.2.3 MAPLHGR and Maximum LHGR Operating Limits

CPPU operating conditions do not usually affect the MAPLHGR or Maximum LHGR Operating Limits. The MAPLHGR and Maximum LHGR Operating Limits ensure that the plant does not exceed regulatory limits established in 10 CFR 50.46 or by the fuel design limits. The MAPLHGR Operating Limit is determined by analyzing the limiting LOCA for the plant. As discussed in Section 4.3 of the CLTR, [[ ]] The Maximum LHGR Operating Limit is determined by the fuel rod thermal-mechanical design and is not affected by CPPU. [[

]]

2.3 REACTIVITY CHARACTERISTICS

The effect of CPPU on shutdown margin and hot excess reactivity for VYNPS is described below. The topics addressed in this evaluation are:

Topic	CPPU Disposition	VYNPS Result
Hot excess reactivity	[[	
Shutdown margin		]]

The general effect of power uprate on core reactivity is described in Section 5.7.1 of ELTR1, and is also applicable for a CPPU. Based on experience with previous plant-specific power uprate submittals, the required hot excess reactivity and shutdown margin can be achieved for CPPU

through appropriate fuel and core design. [[ ]]plant shutdown and reactivity margins must meet NRC approved limits established in Reference 5 on a cycle specific basis and are evaluated for each plant reload core, [[

]]

2.4 STABILITY

Section 3.2 of ELTR1 documents interim corrective actions and four long-term stability options. VYNPS has adopted Option I-D. Option I-D evaluations are core reload dependent and are performed for each reload fuel cycle.

Topic	CPPU Disposition	VYNPS Result
2.4.1 Enhanced Option I-A	[[	
2.4.2 Option I-D		
2.4.3 Option II		
2.4.4 Option III (OPRM armed region and trip setpoint)		
2.4.4 Option III (Hot channel oscillation magnitude)		]]

2.4.1 Plants with Enhanced Option I-A

Not applicable to VYNPS.

2.4.2 Plants with Option I-D

Option I-D is a solution combining prevention and detect-and-suppress elements. The prevention portion of the solution is an administratively controlled exclusion region. The exclusion region calculation is a confirmation that a regional mode instability is not probable. The flow-biased APRM scram provides automatic detection and suppression of core wide instabilities. This scram ensures that the Fuel Cladding Integrity Safety Limit is met for thermal hydraulic oscillations.

CPPU will affect the Exclusion Region. However, the Exclusion Region is dependent upon the core loading, and is reviewed and adjusted, as required, for each reload core. The confirmation that regional mode reactor instability is not probable is also re-evaluated when the Exclusion Region is recalculated. [[ ]] these features will be analyzed for [[ ]] the new rated power level.

CPPU also affects the SLMCPR protection confirmation. Changes to the nominal flow-biased APRM trip setpoint or the rated rod line require the hot bundle oscillation magnitude portion of the detect-and-suppress calculation to be recalculated. This calculation is not dependent upon the core and fuel design. However, the SLMCPR protection calculation is dependent upon the core and fuel design and is performed for each reload. [[ ]] these features will be analyzed for [[ ]] the new rated power level. [[ ]]

An additional analysis was performed to support the development of the CPPU APRM flow-biased scram setpoints contained in Section 5.3.3. The MELLLA analysis (Reference 6) developed a stability-based APRM flow-biased scram equation that defines percent CLTP (P) in terms of percent rated core flow (W_c). The slope and intercept of the Reference 6 APRM flow-biased equation were rescaled for CPPU so that the absolute value of P, in terms of thermal power (MW_t) versus core flow, is unchanged from CLTP to CPPU. The rescaled equation for CPPU is: $1.3501W_c + 11.40$.

For stability Option I-D, the dominance of the core-wide mode oscillation can be demonstrated by calculating the channel decay ratio at the most limiting power / flow state point. This power / flow state point for CPPU is at 49.4% of CPPU RTP / 31.3% of rated core flow; which is identical to the Reference 6 power / flow state point in absolute core thermal power. As stated in Reference 6, [[

]] Therefore, the Option I-D solution remains valid for VYNPS CPPU operation. Thus, the CPPU stability-based APRM flow-biased equation is appropriate for the determination of the APRM flow-biased scram equation in Section 5.3.3. The CPPU APRM flow-biased scram equation will be confirmed as applicable for the VYNPS stability Option I-D solution in the plant-specific reload licensing prior to EPU implementation. This is consistent with Section 2.4 of the CLTR.

2.4.3 Plants with Option II

Not applicable to VYNPS.

2.4.4 Plants with Option III

Not applicable to VYNPS.

2.5 REACTIVITY CONTROL

The Control Rod Drive (CRD) system is used to control core reactivity by positioning neutron absorbing control rods within the reactor and to scram the reactor by rapidly inserting withdrawn control rods into the core. No change is made to the control rods due to the CPPU. The effect on the

nuclear characteristics of the fuel is discussed in Section 2.3. The topics addressed in this evaluation for VYNPS are:

Topic	CPPU Disposition	VYNPS Result
2.5.1 Scram Time Response	[[	
2.5.2 CRD Positioning		
2.5.2 CRD Cooling		
2.5.3 CRD Integrity		]]

2.5.1 Control Rod Scram

For pre-BWR6 plants, the scram times are decreased by the increased transient pressure response, [[ ]] At normal operating conditions, the CRD Hydraulic Control Unit accumulator supplies the initial scram pressure and, as the scram continues, the reactor becomes the primary source of pressure to complete the scram. [[

]]

2.5.2 Control Rod Drive Positioning and Cooling

[[ ]] and the automatic operation of the system flow control valve maintains the required drive water pressure and cooling water flow rate. Therefore, the CRD positioning and cooling functions are not affected. The CRD cooling and normal CRD positioning functions are operational considerations, not safety-related functions, and are not affected by CPPU operating conditions.

Plant operating data has confirmed that the CRD system flow control valve operating position has sufficient operating margin. [[

]]

2.5.3 Control Rod Drive Integrity Assessment

The postulated abnormal operating condition for the CRD design assumes a failure of the CRD system pressure-regulating valve that applies the maximum pump discharge pressure to the CRD mechanism internal components. This postulated abnormal pressure bounds the ASME reactor overpressure limit. [[

]]

Other mechanical loadings are addressed in Section 3.3.2 of this report.

[[

]]

3. REACTOR COOLANT AND CONNECTED SYSTEMS

This section primarily focuses on the information requested in RG 1.70, Chapter 5, and to a very limited extent Chapter 3, that applies to CPPU.

3.1 NUCLEAR SYSTEM PRESSURE RELIEF/OVERPRESSURE PROTECTION

The nuclear system pressure relief system topics addressed in this evaluation are as follows:

Topic	CPPU Disposition	VYNPS Result
Overpressure capacity	[[	
Flow-induced vibration		]]

The nuclear system pressure relief system prevents overpressurization of the nuclear system during AOOs, the plant ASME Upset overpressure protection event, and postulated ATWS events. The plant SRVs and Spring Safety Valves (SSVs) along with other functions provide this protection. An evaluation was performed in order to confirm the adequacy of the pressure relief system for CPPU conditions. The adequacy of the pressure relief system is also demonstrated by the overpressure protection evaluation performed for each reload core and by the ATWS evaluation performed for CPPU (Section 9.3.1).

For VYNPS, no SRV or SSV setpoint increase is needed because there is no change in the dome pressure or simmer margin. Therefore, there is no effect on valve functionality (opening / closing).

Two potentially limiting overpressure protection events are typically analyzed for CPPU: (1) Main Steam Isolation Valve Closure with Scram on High Flux (MSIVF) and (2) Turbine Trip with Bypass Failure and Scram on High Flux (ELTR1, Section 5.5.1.4). However, based on both plant initial core analyses and subsequent power uprate evaluations, the MSIVF is more limiting than the Turbine Trip (TT) event with respect to reactor overpressure. Recent EPU evaluations show a 24 to 40 psi difference between these two events. Only the MSIVF event was performed because it is limiting. In addition, an evaluation of the MSIVF event is performed with each reload analysis.

The design pressure of the reactor vessel and Reactor Coolant Pressure Boundary (RCPB) remains at 1250 psig. The acceptance limit for pressurization events is the ASME code allowable peak pressure of 1375 psig (110% of design value). The overpressure protection analysis description and analysis method are provided in ELTR1. The MSIVF event is conservatively analyzed assuming a failure of the valve position scram. The analyses also assume that the event initiates at a reactor dome pressure of 1040 psia (which is higher than the nominal CPPU dome pressure), and one SRV OOS. Starting from 102% of CPPU RTP, the calculated peak Reactor Pressure Vessel (RPV) pressure, located at the bottom of the vessel, is 1328 psig. The corresponding calculated maximum reactor dome pressure is 1304 psig. The peak calculated RPV pressure remains below the 1375-psig

ASME limit, and the maximum calculated dome pressure remains below the Technical Specification 1335 psig Safety Limit. Therefore, the results are acceptable. The results of the CPPU overpressure protection analysis for the VYNPS MSIVF event are consistent with the generic analysis in ELTR2. The VYNPS response to the MSIVF event is provided as Figure 3-1.

The Main Steam Isolation Valve Closure with Direct Scram event (MSIVD) was analyzed to demonstrate adequate margin to the lifting of unpiped SSVs for VYNPS at CPPU conditions. This analysis shows a pressure margin of 88.9 psi, which exceeds the recommended criterion of 60 psi.

SRV setpoint tolerance is independent of CPPU. CPPU evaluations are performed using the existing SRV setpoint tolerance analytical limit of 3% as a basis. Actual historical in-service surveillance of SRV setpoint performance test results are monitored separately for compliance to the Technical Specification requirements.

The in-service surveillance testing of the plant's SRVs have not shown a significant propensity for high setpoint drift greater than 3%. Out of 25 SRV tests, from the "as found" setpoint lift verification tests performed from 1992 to 2001, only one SRV was found to exceed its setpoint by greater than ± 3%.

Flow-induced vibration (FIV) may increase incidents of valve leakage. However, VYNPS currently has procedures to address a leaking SRV. FIV on the Target Rock 3-Stage safety/relief design may result in an inadvertent SRV opening and a "stuck open" SRV event. This characteristic has previously been identified and is addressed in plant procedures. The consequences of a stuck open SRV have been previously considered in the plant-specific safety analyses and have been demonstrated to be non-limiting.

Increased main steam line flow may affect FIV of the piping and safety/relief valves during normal operation. The vibration frequency, extent and magnitude depend upon plant-specific parameters, valve locations, the valve design and piping support arrangements. The FIV of the piping will be addressed by vibration testing during initial plant operation at the higher steam flow rates (see Sections 3.4.1 and 10.4).

[[

]]

3.2 REACTOR VESSEL

The RPV structure and support components form a pressure boundary to contain the reactor coolant and moderator, and form a boundary against leakage of radioactive materials into the drywell. The RPV also provides structural support for the reactor core and internals. The topics addressed in this evaluation are:

Topic	CPPU Disposition	VYNPS Result
3.2.1 Fracture Toughness	[[	
3.2.2 Reactor Vessel Structural Evaluation (Components not significantly affected)		
3.2.2 Reactor Vessel Structural Evaluation (Affected components)		]]

3.2.1 Fracture Toughness

The CLTR, Section 3.2.1 describes the RPV fracture toughness evaluation process. RPV embrittlement is caused by neutron exposure of the wall adjacent to the core (the "beltline" region). Operation at the CPPU conditions results in a higher neutron flux, which increases the integrated fluence over the period of plant life.

The neutron fluence for both pre-CPPU and CPPU was recalculated using two-dimensional neutron transport theory (Reference 7); the neutron transport methodology is consistent with RG 1.190. The revised fluence is used to evaluate the vessel against the requirements of 10 CFR 50, Appendix G. The results of these evaluations indicate that:

(a) The Upper Shelf Energy (USE) remains bounded by the BWR Owners Group (BWROG) equivalent margin analysis, thereby demonstrating compliance with Appendix G. The results of this analysis are provided in Table 3-1 including a summary of the surveillance capsule, CLTP, and CPPU values using the RG 1.190 consistent fluence.

(b) The beltline material reference temperature of the nil-ductility transition (RT_{NDT}) remains below 200°F.

(c) The pressure versus temperature curves contained in the Technical Specifications remain bounding.

(d) The 33 Effective Full Power Year (EFPY) shifts are decreased, and consequently, result in a change in the adjusted reference temperature, which is the initial RT_{NDT} plus the shift. These values are provided in Table 3-2 along with the CLTP values.

(e) The surveillance program consists of three capsules. One capsule containing Charpy specimens was removed from the vessel after 7.54 EFPY of operation and tested. The remaining two capsules have been in the reactor vessel since plant startup. VYNPS

anticipates participation in the BWR Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (LAR submitted via Reference 8) upon receipt of the NRC license amendment (expected approximately March 2004) and will comply with the withdrawal schedule specified for the surrogate surveillance capsules that now represent VYNPS. CPPU has no effect on the existing surveillance schedule.

The maximum normal operating dome pressure for CPPU is unchanged from that for original power operation. Therefore, the hydrostatic and leakage test pressures are acceptable for the CPPU. Because the vessel is still in compliance with the regulatory requirements, operation with CPPU does not have an adverse effect on the reactor vessel fracture toughness.

3.2.2 Reactor Vessel Structural Evaluation

[[

]]

The effect of CPPU was evaluated to ensure that the reactor vessel components continue to comply with the existing structural requirements of the ASME Boiler and Pressure Vessel Code. For the components under consideration, the 1965 code edition with addenda to and including summer 1966, which is the code of construction, was used as the governing code. However, if a component's design has been modified, the governing code for that component was the code used in the stress analysis of the modified component. The following components were modified since the original construction of VYNPS:

- The Core Spray (CS) nozzle and safe end were modified and the governing code for the modification is ASME Code, Section XI, Subsection IWB-3641, 1983 edition with addenda through winter 1986.
- The recirculation system inlet and outlet safe ends were modified and the governing code for the modification is ASME Code, Section III, 1980 edition with addenda through summer 1982.
- The instrument line nozzle N11 and N12 safe ends were modified and the governing code for the modification is ASME Code, Section III, 1980 edition with addenda through summer 1982.

- The CRD Return Nozzle was decommissioned (capped) and the governing code for the modification is ASME Code, Section III, 1977 edition.

Typically, new stresses are determined by scaling the "original" stresses based on the CPPU conditions (pressure, temperature, and flow). The analyses were performed for the design, the normal and upset, and the emergency and faulted conditions. If there is an increase in annulus pressurization, jet reaction, pipe restraint, or fuel lift loads, the changes are considered in the analysis of the components affected for Normal, Upset, Emergency, and Faulted conditions.

3.2.2.1 Design Conditions

Because there are no changes in the design conditions due to CPPU, the design stresses are unchanged and the Code requirements are met.

3.2.2.2 Normal and Upset Conditions

The reactor coolant temperature and flows at CPPU conditions are only slightly changed from those at current rated conditions. Evaluations were performed at conditions that bound the slight change in operating conditions. The type of evaluations is reconciliation of the stresses and usage factors to reflect CPPU conditions. A primary plus secondary stress analysis was performed showing CPPU stresses still meet the requirements of the ASME Code, Section III, Subsection NB. Lastly, the fatigue usage was evaluated for the limiting location of components with a usage factor greater than 0.5. The VYNPS fatigue analysis results for the limiting components are provided in Table 3-3. The VYNPS analysis results for CPPU show that all components meet their ASME Code requirements.

For the FW nozzle blend radius location, in addition to a stress and fatigue analysis, a fracture mechanics analysis was used in conjunction with inner surface exams and cycle counting to assure potential crack growth is smaller in relation to ASME XI limits. The Ultrasonic Testing (UT) inspection of the inner surface of the FW nozzles is based on a BWROG report (Reference 9) that was approved by the NRC (Reference 10) as an alternative to NUREG-0619. VYNPS also uses thermocouples attached to the outer surface of each FW nozzle to monitor the inconel thermal sleeve interference fit.

The fracture mechanics analysis evaluates crack growth for conservative design transients. Design cycles are then monitored through plant procedure. The conservative design transients used in the fracture mechanics evaluation conservatively bound changes under CPPU conditions. The thermal model used in this assessment employed heat transfer coefficients and a temperature profile that remains conservative under CPPU conditions. Therefore, cycle limits and inspection frequency are not affected by CPPU conditions.

3.2.2.3 Emergency and Faulted Conditions

The stresses due to Emergency and Faulted conditions are based on loads such as peak dome pressure, which are unchanged. Therefore, Code requirements are met for all RPV components.

3.3 REACTOR INTERNALS

The reactor internals include Core Support Structure (CSS) and non-CSS components. The topics addressed in this section are:

Topic	CPPU Disposition	VYNPS Result
3.3.1 Reactor Internals Pressure Differences	[[	
3.3.2 Reactor Internals Structural Evaluation		
3.3.3 Steam Dryer Separator Performance		]]

3.3.1 Reactor Internal Pressure Differences

The increase in core average power alone would result in higher core loads and RIPDs due to the higher core exit steam quality. The maximum acoustic and flow-induced loads, following a postulated recirculation line break, occur at an initial condition that maximizes the reactor vessel downcomer annulus subcooling. The MELLLA analysis (Reference 6) showed that, for VYNPS, the most limiting subcooling condition is at the intersection of the minimum pump speed and the MELLLA flow control line. This limiting condition is also applicable to the VYNPS CPPU because there is no change in the MELLLA flow control line. The vessel downcomer annulus subcooling at the 1.02% of CPPU RTP and 107% of rated core flow (ICF) condition remains lower than at the limiting condition. Therefore, the maximum acoustic and flow-induced loads, following a postulated recirculation line break, are unaffected by the CPPU.

The RIPDs are calculated for Normal (steady-state operation), Upset, and Faulted conditions for all major reactor internal components. For VYNPS, the Emergency condition RIPDs are bounded by the Faulted condition RIPDs, because the RPV depressurization rate for the Emergency condition event (inadvertent actuation of the Automatic Depressurization System (ADS), assuming the maximum power/maximum core flow point) is slower and results in lower RIPDs than the Faulted events (Main Steam Line Break (MSLB) inside or outside containment).
[[

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The following assumptions and initial conditions were used in the determination of the Normal, Upset, and Faulted condition RIPDs for CPPU RTP operation.

Condition	Limiting Event	Initial Conditions	Bases/Justifications
Normal	Steady State Operation	102%P / 107%F	Maximum power and flow during normal operation. Consistent with existing VYNPS licensing basis
Upset	N/A	102%P / 107%F	Maximum power and flow attainable during anticipated plant transients. Consistent with existing VYNPS licensing basis.
Faulted	MSLB Inside Containment	• High Power: 102%P / 107%F • Low Power / High Flow: 16.7%P / 110%F [1.]	Design basis MSLB (limiting transient event), which results in the most challenging plant condition (maximum loads on the local components). The high power point is used because the maximum loads occur at the maximum core flow and maximum void formation in the bundles. The low power/high flow point is used because it results in a higher mismatch between the steam flow from the break and the steam generated in the core.
Faulted	MSLB outside Containment	• Low Power / High Flow: 16.7%P / 110%F [1.]	Limiting condition for steam dryer pressure drop. The steam dryer pressure drop at the low power high flow condition bounds (is higher than) the steam dryer pressure drop at high power, due to the larger mismatch in fluid flow.

Note:

1. The High Flow value (110%F) for the Faulted condition is consistent with the ICF evaluation (Reference 11).

Tables 3-4 through 3-6 compare results for the various loading conditions between the current analysis results and operation with CPPU for the vessel internals that are affected by the changed RIPDs.

3.3.2 Reactor Internals Structural Evaluation

The reactor internals consist of the CSS components and non-core support structure (non-CSS) components. The RPV internals (excluding the CRD) are not certified to the ASME code; however, the requirements of the ASME Code were used as guidelines in their design basis analysis. The evaluations/stress reconciliation in support of the CPPU was performed consistent with the design basis analysis of the components. The reactor internal components evaluated in this section are:

- Shroud
- Shroud Support
- Core Plate
- Top Guide
- CRD Housing and CRD
- Control Rod Guide Tube
- Orificed Fuel Support
- Fuel Channel
- Steam Dryer
- Feedwater Sparger
- Jet Pumps
- CS Line and Sparger
- Access Hole Cover
- Shroud Head and Steam Separator Assembly (including shroud head bolts)
- In-core Housing and Guide Tube

The original configuration of the reactor internals is considered in the CPPU evaluation unless a component has undergone permanent structural modifications, in which case, the modified configuration is used as the basis for the evaluation. The following components have permanent structural modifications from the original configuration: FW sparger replacement, CS sparger repair, core plate plugging, and shroud repair.

The effects on the loads as a result of the thermal-hydraulic changes due to CPPU are evaluated for the reactor internals. All applicable Normal, Upset, Emergency, and Faulted service condition loads and load combinations are considered consistent with the existing design basis analysis. These loads include the RIPDs, seismic loads, flow induced and acoustic loads due to Recirculation Line Break - Loss-of-Coolant Accident (RLB-LOCA), fuel lift loads, and thermal loads. The RIPDs increase for some components as a result of CPPU. The flow conditions and thermal effects were considered in the evaluation, as applicable. The seismic response is

unaffected by CPPU, and the fuel lift loads for CPPU are the same as those of the CLTP. The acoustic and flow induced loads due to RLB-LOCA remain unaffected relative to CLTP.

A qualitative or quantitative assessment was performed for the RPV internals consistent with the existing design basis and the severity of the load change. The CPPU loads are compared to those in the existing design basis analysis. If the loads do not increase due to CPPU, then the existing analysis results bound the CPPU conditions, and no further evaluation is required or performed. If the loads increase due to the CPPU, then the effect of the load increase is evaluated further. [[

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Table 3-7 presents the governing stresses for the various reactor internal components of VYNPS as affected by CPPU. All stresses are within allowable limits, and the RPV internals are demonstrated to be structurally adequate for operation in the CPPU condition.

The following reactor vessel internals were evaluated for the effects of changes in loads due to CPPU.

(a) **Shroud and Shroud Repair:** The core shroud and shroud repair components were evaluated for the loads defined for the CPPU condition. All loads were considered including differential pressure loads across the reactor internals structures for Normal, Upset, and Faulted conditions, seismic loads, recirculation line break loads (both flow-induced and acoustic), and plant transient loads. For the shroud repair, the evaluations included the components that carry the vertical loads (the spring rod and adapters and the connections to the shroud flange and shroud support plate) as well as the components that carry the lateral loads (the lateral restraints). The evaluation also addressed the loads and stresses in the interfacing core shroud and reactor vessel structures.

The loads for the CPPU condition were assessed by revising or updating, as appropriate, the evaluations previously performed for the CLTP condition. The NRC Staff reviewed the VYNPS shroud repair modification and found it acceptable (Reference 12). The results of the CPPU evaluations showed that the loads and stresses for all shroud repair components and interfacing core shroud and reactor vessel structures remain within design allowables.

In Table 3-6, the allowable loads are compared to the applied loads for the CLTP and CPPU conditions for the limiting shroud repair component. As shown, the allowable load exceeds the CLTP and CPPU loads with substantial remaining margin.

In addition to the structural evaluations that are summarized above, evaluations were performed to confirm that the current shroud repair installation preload is adequate for the CPPU loads. This evaluation confirmed that the preload is sufficient to maintain a compressive load in the shroud for all normal plant operating conditions and thereby

sufficient to prevent separation of shroud sections even in the event of through-wall, 360° cracking in all of the shroud horizontal welds.

(b) **Shroud Support**: An evaluation of the shroud support plate was performed for the CPPU loads. As discussed in Section 3.3.2 (a) above, the shroud support plate serves as the attachment location for the vertical shroud repair members and therefore transmits the vertical load from the shroud repair to the reactor vessel. In Table 3-6, the allowable loads for the shroud support plate are compared to the applied loads for the CLTP and CPPU conditions. This comparison shows that the shroud support plate has sufficient capability to transmit the applied loads with substantial remaining margin.

(c) **Core Plate: [[**

]]

Therefore, the core plate remains structurally qualified for CPPU.

(d) **Top Guide: [[**

]] Therefore, the structural integrity of the Top Guide is maintained for CPPU.

(e) **CRD Housing and CRD: [[**

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Therefore, the structural integrity of the CRD housing is maintained for CPPU.

The CRD, which is inside the CRD housing, contains ASME Code components. [[

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Therefore, the structural integrity of the CRD is maintained for CPPU.

(f) **Control Rod Guide Tube: [[**

]] Therefore, the structural integrity of the Control Rod Guide Tube is maintained for CPPU.

(g) **Orificed Fuel Support:** [[

]] Therefore, the structural integrity of the OFS is maintained for CPPU.

(h) **Fuel Channels:** [[
]] Therefore, the structural integrity of the fuel channels is maintained for CPPU.

(i) **Steam Dryer:** [[

]] Therefore, the structural integrity of the steam dryer is maintained for CPPU.

(j) **Feedwater Sparger:** [[

]] Therefore, the structural integrity of the feedwater sparger is maintained for CPPU.

(k) **Jet Pumps:** [[

]] the structural integrity of the jet pump assembly is maintained for CPPU.

(l) **CS Lines and Sparger:** [[

]] Therefore, the structural integrity of the CS line and spargers is maintained for CPPU.

(m)Access Hole Cover: [[

]] Therefore, the structural integrity of the access hole cover is maintained for CPPU.

(n) Shroud Head and Steam Separator Assembly (including Shroud Head Bolts): [[

]] Therefore, the structural integrity of the shroud head and steam separator assembly is maintained for CPPU.

(o) In-core Housing and Guide Tube: [[

]] Therefore, the structural integrity of the in-core housing and guide tube is maintained for CPPU.

3.3.3 Steam Dryer/Separator Performance

At VYNPS, the performance of the steam separators and dryer has been evaluated to ensure that the quality of the steam leaving the reactor pressure vessel continues to meet existing operational criteria at CPPU conditions. CPPU results in an increase in saturated steam generated in the reactor core. For constant core flow, this in turn results in an increase in the separator inlet quality and dryer face velocity and a decrease in the water level inside the dryer skirt. These factors, in addition to the radial power distribution affect the steam separator-dryer performance. The results of the evaluation demonstrate that the steam separator-dryer performance remains acceptable (e.g., moisture content ≤ 0.1 weight %) at CPPU conditions.

3.4 FLOW INDUCED VIBRATION

The FIV evaluation addresses the influence of an increase in flow during CPPU on RCPB piping, RCPB piping components, and RPV internals. The topics addressed in this evaluation are:

Topic	CPPU Disposition	VYNPS Result
3.4.1 Structural Evaluation of Recirculation Piping	[[	
3.4.1 Structural Evaluation of Main Steam and Feedwater Piping		
3.4.1 Safety-Related Thermowells and Probes		
3.4.2 Structural Evaluation of core flow dependent RPV Internals		
3.4.2 Structural Evaluation of other RPV Internals		]]

3.4.1 FIV Influence on Piping

Key applicable structures include the Main Steam (MS) system piping and suspension, the Feedwater (FW) system piping and suspension, and the Reactor Recirculation System (RRS) system piping and suspension. In addition, branch lines attached to the MS system piping or FW system piping are considered.

RRS drive flow is not significantly increased (1.9%) during CPPU operation. [[

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The MS and FW piping have increased flow rates and flow velocities in order to accommodate CPPU. As a result, the MS and FW piping experience increased vibration levels, approximately proportional to the square of the flow velocities. The ASME Code (NB-3622.3) and nuclear regulatory guidelines require some vibration test data be taken and evaluated for these high energy piping systems during initial operation at CPPU conditions. Vibration data for the MS and FW piping inside containment will be acquired using remote sensors, such as displacement probes, velocity sensors, and accelerometers. A piping vibration startup test program, consistent with the ASME code and regulatory requirements, will be performed.

[[

]] and FIV testing of the MS and FW piping system will be performed during CPPU power ascension.

There are no safety-related thermowells and sample probes in the MS and FW piping systems at VYNPS. Therefore, no evaluation was required for CPPU. The non-safety related thermowells and sample probes will be evaluated during the piping vibration startup test program.

3.4.2 FIV Influence on Reactor Internal Components

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The required reactor vessel internals vibration assessment of the other RPV internals is described in the CLTR. CPPU operation increases the steam production in the core, resulting in an increase in the core pressure drop. There is only a slight increase (1.9%) in maximum drive flow at CPPU conditions for VYNPS as compared to CLTP. The increase in power may increase the level of reactor internals vibration. Analyses were performed to evaluate the effects of FIV on the reactor internals at CPPU conditions. This evaluation used a bounding reactor power of 1912 MWt and 107% of rated core flow. This assessment was based on vibration data obtained during startup testing of the prototype plant (Monticello). For components requiring an evaluation but not instrumented in the prototype plant, vibration data acquired during the startup testing from similar plants or acquired outside the RPV is used. The expected vibration levels for CPPU were estimated by extrapolating the vibration data recorded in the prototype plant or similar plants and on GE BWR operating experience. These expected vibration levels were then compared with the established vibration acceptance limits. The following components were evaluated:

- Shroud head and separator
- Jet pumps
- Feedwater sparger
- In-core guide tubes (generic disposition)
- Control rod guide tubes (generic disposition)
- Steam dryer
- Jet pump sensing lines

The results of the vibration evaluation show that continuous operation at a reactor power of 1912 MWt and 107% of rated core flow does not result in any detrimental effects on the safety-related reactor internal components.

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During CPPU, the components in the upper zone of the reactor, such as the moisture separators and dryer, are mostly affected by the increased steam flow. Components in the core region and components such as the CS line are primarily affected by the core flow. Components in the annulus region such as the jet pump are primarily affected by the recirculation pump drive flow and core flow. Because there is a slight increase (1.9%) in maximum drive flow with core flow remaining the same as compared to the CLTP condition, small increase in FIV on the components in the annular and core regions are expected. However, the steam separator and dryer are significantly affected by CPPU conditions.

The steam dryer and steam separators are non safety-related components. Recent uprate experience indicates that FIV at CPPU conditions may lead to high cycle fatigue failure of some dryer components. Failure of a dryer component does not represent a safety concern, but can result in a large economic impact. A qualitative evaluation of the VYNPS steam dryer has been performed, with resulting preliminary modifications and inspections identified to enhance dryer structural integrity at CPPU conditions. The preliminary modifications include replacing or reinforcing the steam dryer cover plates.

A quantitative evaluation is being performed to identify dryer components susceptible to failure at CPPU conditions. The results of the quantitative evaluation will be used to finalize the modifications needed to maintain steam dryer structural integrity at CPPU conditions. Any identified dryer modifications will be performed prior to CPPU implementation.

The calculations for CPPU conditions indicate that vibrations of all safety-related reactor internal components are within the GE acceptance criteria. The analysis is conservative for the following reasons:

- The GE criteria of 10,000 psi peak stress intensity is less than the ASME Code criteria of 13,600 psi;
- The modes are absolute summed; and
- The maximum vibration amplitude in each mode is used in the absolute sum process, whereas in reality the peak vibration amplitudes are unlikely to occur at the same time.

Based on the above, it is concluded that FIV effects are expected to remain within acceptable limits at CPPU conditions.

3.5 PIPING EVALUATION

3.5.1 Reactor Coolant Pressure Boundary Piping

The RCPB piping systems evaluation consists of a number of safety-related piping subsystems that move fluid through the reactor and other safety systems. The topics addressed in this evaluation are:

Topic	CPPU Disposition	VYNPS Result
Structural evaluation for unaffected safety-related piping	[[	
Structural evaluation for affected safety-related piping		]]

The flow, pressure, temperature, and mechanical loading for most of the RCPB piping systems do not increase for CPPU. [[

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The following piping system segments from the RPV to the normally closed containment isolation valve are [[

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Section 3.1 demonstrates that the RCPB piping remains below the ASME pressure limit during the most severe pressurization transient.

For VYNPS, the plant-specific piping evaluation process is consistent with the methodology described in Appendix K of ELTR1, which has been reviewed and accepted by the NRC for power uprate evaluations. This process involves comparing existing piping data (i.e., temperatures, pressures, and flow rates) with the corresponding data at CPPU conditions to determine piping

system acceptability. Existing piping stresses and pipe support loads are increased, as required, to evaluate CPPU conditions. These revised stresses and pipe support loads were evaluated and are within acceptable design limits.

Main Steam and Associated Piping System Evaluation

The Main Steam (MS) piping system and associated branch piping (inside containment) were evaluated for compliance with the USAS-B31.1.0-1967 Power Piping Code stress criteria for the effects of CPPU on piping, piping supports including the associated building structure, piping interfaces with the RPV nozzles, penetrations, flanges and valves.

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A bounding piping analysis was performed which included the effects of CPPU along with the additional SSV and larger orifices for the two existing SSV that will be installed as a result of ARTS/MELLLA. The increase in MS flow results in increased forces from the turbine stop valve closure transient. The turbine stop valve closure loads bound the MSIV closure loads because the MSIV closure time is significantly longer than the stop valve closure time.

Pipe Stresses

A review of the increase in flow associated with CPPU indicates that piping load changes do not result in load limits being exceeded for the MS system and attached branch piping or for RPV nozzles. The original design analyses have sufficient margin between calculated stresses and USAS-B31.1.0-1967 Code allowable limits to justify operation at CPPU conditions. The pressure and temperature of the MS piping are unchanged for the CPPU.

Similarly, the branch pipelines (Safety Relief Valve Discharge Line (SRVDL), Reactor Core Isolation Cooling (RCIC), High Pressure Coolant Injection (HPCI), RPV Vent, and MSIV Drain) connected to the MS headers were evaluated to determine the effect of the increased MS flow on the lines. This evaluation concluded that there is no effect on the existing MS branch line qualifications due to the increased flows resulting from CPPU. As with the MS piping, the pressures and temperatures for these branch pipelines do not change as a result of CPPU. No new postulated break locations were identified.

Pipe Supports

The main steam piping (inside containment) was evaluated for the effects of flow increase on the piping snubbers, hangers, struts, and pipe whip restraints. With the exception of the pipe clamps for supports RMSH-6 (MS-35) and RMSH-14 (MS-6), which will be modified, a review of the increase in MS flow associated with CPPU indicates that piping load changes do not result in any load limit being exceeded. Based on existing margins available for the main steam piping supports,

it was concluded that CPPU does not result in reactions on existing structures in excess of the current design capacity.

Feedwater Evaluation

The FW system (inside containment) was evaluated for compliance with the USAS-B31.1.0-1967 Power Piping Code stress criteria for the effects of thermal expansion load and displacements on the piping snubbers, hangers, and struts. Piping interfaces with RPV nozzles, penetrations, flanges and valves were also evaluated.

Pipe Stresses

A review of the small increases in temperature and flow associated with CPPU indicates that piping load changes do not result in load limits being exceeded for the FW piping system or for RPV nozzles. The original design analyses have sufficient design margin between calculated stresses and USAS-B31.1.0-1967 Code allowable limits to justify operation at CPPU conditions.

The design adequacy evaluation shows that the requirements of USAS-B31.1.0-1967 Code requirements remain satisfied. Therefore, CPPU does not have an adverse effect on the FW piping design. No new postulated pipe break locations were identified.

Pipe Supports

The FW system was evaluated for the effects of thermal expansion displacements on the piping snubbers, hangers, and struts. A review of the increases in temperature and FW flow associated with CPPU indicates that piping load changes do not result in any load limit being exceeded. Based on existing margins available for the feedwater piping supports, it was concluded that CPPU does not result in reactions on existing structures in excess of the current design capacity.

Other RCPB Piping Evaluation

This section addresses the adequacy of the other RCPB piping designs, for operation at the CPPU conditions. The nominal operating pressure and temperature of the reactor are not changed by CPPU. Aside from MS and FW, no other system connected to the RCPB experiences an increased flow rate at CPPU conditions. Only minor changes to fluid conditions are experienced by these systems due to higher steam flow from the reactor and the subsequent change in fluid conditions within the reactor.

These systems were evaluated for compliance with the USAS B31.1 or ASME Code stress criteria (as applicable). Because none of these piping systems experience any significant change in operating conditions, they are all acceptable as currently designed.

3.5.2 Balance-Of-Plant Piping

The BOP Piping systems evaluation consists of a number of piping subsystems that move fluid

through systems outside the RCPB piping. The topics addressed in this evaluation are:

Topic	CPPU Disposition	VYNPS Result
Structural evaluation for unaffected non-safety-related piping	[[	
Structural evaluation for affected non-safety-related piping		]]

[[

]] For VYNPS, the plant-specific piping evaluation process is consistent with the methodology described in Appendix K of ELTR1, which has been reviewed and accepted by the NRC for power uprate evaluations. This process involves comparing existing piping data (i.e., temperatures, pressures, and flow rates) with the corresponding data at CPPU conditions in order to determine piping system acceptability. Existing piping stresses and pipe support loads are increased, as required, to evaluate CPPU conditions. These revised stresses and pipe support loads were evaluated and are within acceptable design limits.

Large bore and small bore ASME Class 1, 2, and 3 piping and supports not addressed in Section 3.5.1 were evaluated for acceptability at CPPU conditions. The evaluation of the BOP piping and supports was performed in a manner similar to the evaluation of RCPB piping systems and supports (Section 3.5.1), using applicable ASME Section III, Subsections NC/ND or B31.1 Power Piping Code equations. The original Codes of record (as referenced in the appropriate calculations), Code allowables, and analytical techniques were used and no new assumptions were introduced.

The Design Basis Accident (DBA)-LOCA dynamic loads, including the pool swell loads, vent thrust loads, Condensation Oscillation (CO) loads and chugging loads were originally defined and evaluated for VYNPS. The structures attached to the torus shell, such as piping systems, vent penetrations, and valves are based on these DBA-LOCA hydrodynamic loads. For CPPU conditions, the DBA-LOCA torus shell hydrodynamic loads were re-evaluated and found acceptable and there are no resulting effects on the torus shell attached structures.

The increase in the suppression pool temperature to approximately 195°F during the long term Post LOCA condition has been evaluated for all affected piping systems connected to the torus.

The effects of the CPPU conditions have been evaluated for the following piping systems:

- MS (outside containment)
- Extraction Steam (ES), Heater Vents and Drains
- FW and Condensate
- Reactor Water Cleanup (RWCU) - Outside Containment
- RHR - Outside Containment
- RHR Service Water – Outside Containment
- CS - Outside Containment – Pump Suction / Pump Discharge
- HPCI – Outside Containment
- RCIC – Outside Containment
- Standby Liquid Control System (SLCS) - Outside Containment
- CRD
- Service Water
- Reactor Building Closed Cooling Water
- Turbine Building Closed Cooling Water
- Spent Fuel Cooling
- Standby Gas Treatment
- Off Gas
- Torus Attached Piping including ECCS Suction Strainers

Pipe Stresses

Operation at the CPPU conditions increases stresses on piping and piping system components due to slightly higher operating temperatures and flow rates internal to the pipes. For those systems with analysis, the maximum stress levels were reviewed based on specific increases in temperature pressure and flow rate (see Tables 3-8a, 3-8b, and 3-8c). For those systems that do not require a detailed analysis, pipe routing and flexibility was evaluated and determined to be acceptable. These piping systems have been evaluated and meet the appropriate code criteria for the CPPU conditions, based on the design margins between actual stresses and code limits in the original design. All piping is below the applicable code allowable stress limits. No new postulated pipe break locations were identified.

Pipe Supports

Operation at the CPPU conditions slightly increases the pipe support loadings due to increases in the temperature of the affected piping systems.

The pipe supports of the systems affected by CPPU loading increases (MS, RHR, CS, HPCI, RCIC, FW, and ES) were reviewed to determine if there is sufficient margin to code acceptance criteria to accommodate the increased loadings. This review shows that there is adequate design margin between the original design stresses and code limits of the supports to accommodate the load increase, with the exception of support RCIC-HD63C that requires modification. The conservatisms are introduced by the use of the lowest code allowable for various plant loading conditions, the use of generic enveloping design loads instead of actual loads, and the conservative load application on base plates, anchor bolts, and lugs. This review shows that, in all cases, except for support RCIC-HD63C, the support loads under CPPU conditions are in compliance with appropriate Code criteria. A minor modification will be made prior to CPPU implementation to RCIC-HD63C. Based on existing margins available for the affected BOP piping supports, it was concluded that CPPU does not result in reactions on existing structures in excess of the current design capacity.

Main Steam and Associated Piping System Evaluation (Outside containment)

The MS piping system (outside containment) was evaluated for compliance with VYNPS design criteria. Included in the evaluation were the effects of CPPU on piping stresses, piping supports, and the associated building structure, turbine nozzles, and valves.

Because the MS piping pressures and temperatures outside containment are not affected by CPPU, there was no effect on the analyses for these parameters. The increase in MS flow results in increased forces from the turbine stop valve closure transient. The turbine stop valve closure loads bound the MSIV valve loads because the MSIV closure time is significantly longer than the stop valve closure time. The MS analysis results are provided in Table 3-8c.

Pipe Stresses

A review of the increase in flow associated with CPPU indicates that piping load changes do not result in load limits being exceeded for the main steam piping system outside containment. The original design has sufficient design margin to justify operation at the CPPU conditions. The pressure and temperature of the MS piping are unchanged for CPPU. No new postulated break locations were identified.

Pipe Supports

The pipe supports and turbine nozzles for the MS piping system outside containment were evaluated for the increased loading and movements associated with the turbine stop valve closure transient at CPPU conditions. The evaluations demonstrate that the existing piping supports and turbine nozzles are acceptable and can accommodate the increased loads and movements resulting from CPPU. Based on existing margins available for the outside containment

main steam piping supports, it was concluded that CPPU does not result in reactions on existing structures in excess of the current design capacity.

3.6 REACTOR RECIRCULATION SYSTEM

The RRS evaluation for CPPU addressed the following topics:

Topic	CPPU Disposition	VYNPS Result
System evaluation	[[	
Net positive suction head (NPSH)		
Flow mismatch		
Single loop operation		]]

The CPPU power condition is accomplished by operating along extensions of current rod lines on the power/flow map with no increase in the maximum licensed core flow of 107% of rated. The core reload analyses are performed with the most conservative allowable core flow. The evaluation of the reactor recirculation system performance at CPPU power determines that adequate core flow can be maintained.

The cavitation protection interlock remains the same in terms of absolute flow rates. This interlock is based on subcooling in the external recirculation loop and thus is a function of absolute FW flow rate and FW temperature at less than full thermal power operating conditions. Therefore, the interlock is not changed by CPPU.

VYNPS does not have a recirculation pump flow mismatch Technical Specification. [[

]]

SLO operation is limited to off rated conditions and is not affected as a result of the CPPU.

[[

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3.7 MAIN STEAM LINE FLOW RESTRICTORS

The main steam line flow restrictor evaluation for CPPU at VYNPS addressed the following topic:

Topic	CPPU Disposition	VYNPS Result
Structural integrity	[[	]]

The increase in steam flow rate has no significant effect on flow restrictor erosion. There is no effect on the structural integrity of the main steam flow element (restrictor) due to the increased differential pressure because the restrictors were designed and analyzed for the choke flow condition.

After a postulated steam line break outside containment, the fluid flow in the broken steam line increases until it is limited by the main steam line flow restrictor. [[

]] The VYNPS restrictors were originally designed and analyzed for these flow conditions and therefore the restrictors remain within the acceptable calculated differential pressure drop and choke flow limits under CPPU conditions.

3.8 MAIN STEAM ISOLATION VALVES

The MSIV evaluation for CPPU at VYNPS addressed the following topics:

Topic	CPPU Disposition	VYNPS Result
Isolation performance	[[	
Valve pressure drop		]]

The MSIVs are part of the RCPB, and perform the safety function of steam line isolation during certain abnormal events and accidents. The MSIVs must be able to close within a specified time range at all design and operating conditions. They are designed to satisfy leakage limits set forth in the plant Technical Specifications.

The MSIVs have been evaluated, as discussed in Section 4.7 of ELTR2, Supplement 1. The evaluation covers both the effects of the changes to the structural capability of the MSIV to meet pressure boundary requirements, and the potential effects of CPPU-related changes to the safety functions of the MSIVs. The generic evaluation from ELTR2 is based on (1) a 20% thermal power increase, (2) an increased operating dome pressure to 1095 psia, (3) a reactor temperature increase to 556°F, and (4) steam and feedwater increases of about 24%. The evaluation from ELTR2 is confirmed applicable to VYNPS. An increase in flow rate assists MSIV closure, which results in a slightly faster MSIV closure time. The self-compensating feature of the hydraulic control valve will maintain the closing time with little deviation despite the flow rate change. Therefore, CPPU described herein is bounded by conclusions of the evaluation in Section 4.7 of ELTR2, and the MSIVs are acceptable for CPPU operation.

3.9 REACTOR CORE ISOLATION COOLING/ISOLATION CONDENSER

The Isolation Condenser is not applicable to VYNPS.

The RCIC system evaluation for CPPU at VYNPS addressed the following topics:

Topic	CPPU Disposition	VYNPS Result
System performance and hardware	[[	
Net positive suction head		
Adequate core cooling for limiting LOFW events		
Inventory makeup - Operational Level 1 avoidance (For VYNPS, ECCS actuation is initiated at "low-low" reactor water level)		]]

The RCIC system is required to maintain sufficient water inventory in the reactor to permit adequate core cooling following a reactor vessel isolation event accompanied by loss of flow from the FW system. The system design injection rate must be sufficient for compliance with the system limiting criterion to maintain the reactor water level above Top of Active Fuel (TAF) at the CPPU conditions. The RCIC system is designed to pump water into the reactor vessel over a wide range of operating pressures. As described in Section 9.1.3, this event is addressed on a plant-specific basis. The results of the VYNPS plant-specific evaluation indicate adequate water level margin above TAF at the CPPU conditions. Thus, the RCIC injection rate is adequate to meet this design basis event.

An operational requirement is that the RCIC system can restore the reactor water level while avoiding ADS timer initiation and MSIV closure activation functions associated with the low-low reactor water level setpoint. This requirement is intended to avoid unnecessary initiations of safety systems. The results of the VYNPS plant-specific evaluation indicate that the RCIC system is capable of maintaining the water level outside the shroud above the nominal low-low reactor water level setpoint through a limiting Loss of Feedwater Flow (LOFW) event at the CPPU conditions. Thus, the RCIC injection rate is adequate to meet the requirements for inventory makeup. (see Section 9.1.3)

For the CPPU, there is no change to the normal reactor operating pressure and the SRV/SSV setpoints remain the same. There is no change to the maximum specified reactor pressure for RCIC system operation, [[

]] there are no physical changes to the pump suction configuration, and no changes to the system flow rate or minimum atmospheric pressure in the suppression chamber or Condensate Storage Tank (CST). CPPU does not affect the capability to transfer the RCIC pump suction on high suppression pool level or low CST level from its normal alignment, the CST, to the suppression pool, and does not change the existing requirements for the transfer. For ATWS (Section 9.3.1) and fire protection (Section 6.7), operation of the RCIC system at suppression pool temperatures greater than the operational limit may be accomplished by using the dedicated CST volume as the source of water. Therefore, the specified operational temperature limit for the process water does not change with the CPPU. [[

]] The effect of CPPU on the operation of the RCIC system during SBO events is discussed in Section 9.3.2.

The reactor system response to an LOFW transient with RCIC is discussed in Section 9.1.3.

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3.10 RESIDUAL HEAT REMOVAL SYSTEM

The RHR system evaluation for CPPU at VYNPS addressed the following topics:

Topic	CPPU Disposition	VYNPS Result
LPCI mode	[[	
Suppression pool and containment spray cooling modes		
Shutdown cooling mode		
Steam condensing mode		
Fuel pool cooling assist		]]

The RHR system is designed to restore and maintain the reactor coolant inventory following a LOCA and remove reactor decay heat following reactor shutdown for normal, transient, and accident conditions. The CPPU effect on the RHR system is a result of the higher decay heat in the core corresponding to the increased RTP and the increased amount of reactor heat discharged into the containment during a LOCA. For VYNPS, the RHR system is designed to operate in the following modes: LPCI mode, Shutdown Cooling (SDC), Suppression Pool Cooling (SPC), Containment Spray Cooling (CSC), and Fuel Pool Cooling (FPC) assist. The Steam Condensing Mode (SCM) of RHR is not installed at VYNPS.

The LPCI mode, as it relates to the LOCA response, is discussed in Section 4.2.4.

The SPC mode is manually initiated following isolation transients and a postulated LOCA to maintain the containment pressure and suppression pool temperature within design limits. The CSC mode reduces drywell pressure, drywell temperature, and suppression chamber pressure following an accident. The adequacy of these operating modes is demonstrated by the containment analysis (Section 4.1).

The FPC assist mode, using existing RHR heat removal capacity, provides supplemental fuel pool cooling capability in the event that the fuel pool heat load exceeds the heat removal capability of the Fuel Pool Cooling and Cleanup (FPCC) system and the Standby Fuel Pool Cooling System (SFPCS). The adequacy of fuel pool cooling, including use of the FPC assist mode, is addressed in Section 6.3.1.

The higher suppression pool temperature (194.7°F) and containment pressure during a postulated LOCA (Section 4.1) do not affect the hardware capabilities of the RHR equipment, except for the RHR pump seals, to perform the LPCI, SPC, and CSC functions. The peak suppression pool temperature during a limiting LOCA remains below the RHR pump seal design temperature of 210°F. However, this temperature exceeds the maximum operating temperature of 185°F analyzed for the pump seals. Either the pump seals will be re-qualified for the peak suppression pool temperature, or a modification will be completed to ensure seal operation prior to the CPPU implementation.

The effects of CPPU on the remaining modes are discussed in the following subsections.

3.10.1 Shutdown Cooling Mode

[[

]]

3.10.2 Steam Condensing Mode

The SCM is not installed at VYNPS.

3.11 REACTOR WATER CLEANUP SYSTEM

The RWCU system evaluation for CPPU at VYNPS addressed the following topics:

Topic	CPPU Disposition	VYNPS Result
System performance	[[	
Containment isolation		]]

RWCU system operation at the CPPU RTP level slightly decreases the temperature within the RWCU system. This system is designed to remove solid and dissolved impurities from recirculated reactor coolant, thereby reducing the concentration of radioactive and corrosive species in the reactor coolant. The system is capable of performing this function at the CPPU RTP level.

Based on operating experience, the FW iron input to the reactor increases as a result of the increased feedwater flow. This input increases the calculated reactor water iron concentration approximately in proportion to the increase in reactor thermal power. However, this change is considered negligible, and does not affect RWCU.

The effects of CPPU on the RWCU system functional capability have been reviewed, and the system can perform adequately during CPPU with the original RWCU system flow. This RWCU system flow results in a slight increase in the calculated reactor water conductivity (approximately 0.006 μS/cm) because of the increase in FW flow. The present reactor water conductivity limits are unchanged for CPPU and the actual conductivity remains within these limits.

The increase in FW line pressure has a slight effect on the system operating conditions. However, the change due to CPPU does not affect any current analysis in the Generic Letter (GL) 89-10 Program (Section 4.1.4) for the RWCU Motor Operated Valves (MOVs).

Table 3-1a
VYNPS Upper Shelf Energy Equivalent Margin Analysis
40-Year Life (32 EFPY; CLTP)

Plant Applicability Verification Form
for Vermont Yankee
Current Licensing Conditions
40-Year Life (32 EFPY)

BWR/3-6 PLATE

Surveillance Plate USE (Heat C3017):

%Cu	=	0.11	
1st Capsule Fluence	=	4.49E+16 n/cm^2	
1st Capsule Measured % Decrease	=	8.03	(Charpy Curves)
1st Capsule RG 1.99 Predicted % Decrease	=	5.55	(RG 1.99, Rev. 2, Figure 2)
Ratio of Measured to Predicted % Decrease	=	1.448	(RG 1.99, Rev. 2, Position 2.2)

Limiting Beltline Plate USE (Heat C3116):

%Cu	=	0.14	
32 EFPY 1/4T Fluence	=	2.21E+17 n/cm^2	
RG 1.99 Predicted % Decrease	=	9.35	(RG 1.99, Rev. 2, Figure 2)
Adjusted % Decrease	=	13.50	(RG 1.99, Rev. 2, Position 2.2)

13.50% $\leq$ 21%

Therefore, vessel plates are bounded by Equivalent Margin Analysis

Table 3-1b
VYNPS Upper Shelf Energy Equivalent Margin Analysis
40-Year Life (32 EFPY; CLTP)

Equivalent Margin Analysis
Plant Applicability Verification Form
for Vermont Yankee
Current Licensing Conditions
40-Year Life (32 EFPY)

BWR/2-6 WELD

Surveillance Weld USE (SMAW):

%Cu	=	0.03	
1st Capsule Fluence	=	4.49E+16 n/cm^2	
1st Capsule Measured % Decrease	=	4.80	(Charpy Curves)
1st Capsule RG 1.99 Predicted % Decrease	=	4.77	(RG 1.99, Rev. 2, Figure 2)
Ratio of Measured to Predicted % Decrease	=	1.005	(RG 1.99, Rev. 2, Position 2.2)

Limiting Beltline Weld USE (SMAW):

%Cu	=	0.04	
32 EFPY 1/4T Fluence	=	2.21E+17 n/cm^2	
RG 1.99 Predicted % Decrease	=	7.32	(RG 1.99, Rev. 2, Figure 2)
Adjusted % Decrease	=	7.36	(RG 1.99, Rev. 2, Position 2.2)

7.36% $\leq$ 34%

Therefore, vessel welds are bounded by Equivalent Margin Analysis

Table 3-1c
VYNPS Upper Shelf Energy Equivalent Margin Analysis
40-Year Life (33 EFPY; 4.827E+8 MWH)

Equivalent Margin Analysis
Plant Applicability Verification Form
for Vermont Yankee
Including Extended Power Uprate Conditions
40-Year Life (33 EFPY; 4.827E+8 MWH)

BWR/3-6 PLATE

Surveillance Plate USE (Heat C3017):

%Cu	=	0.11	
1st Capsule Fluence	=	4.49E+16 n/cm^2	
1st Capsule Measured % Decrease	=	8.03	(Charpy Curves)
1st Capsule RG 1.99 Predicted % Decrease	=	5.55	(RG 1.99, Rev. 2, Figure 2)
Ratio of Measured to Predicted % Decrease	=	1.448	(RG 1.99, Rev. 2, Position 2.2)

Limiting Beltline Plate USE (Heat C3116):

%Cu	=	0.14	
33 EFPY (4.827E+8 MWH) 1/4T Fluence	=	2.35E+17 n/cm^2	
RG 1.99 Predicted % Decrease	=	9.50	(RG 1.99, Rev. 2, Figure 2)
Adjusted % Decrease	=	13.80	(RG 1.99, Rev. 2, Position 2.2)

13.80% $\leq$ 21%

Therefore, vessel plates are bounded by Equivalent Margin Analysis

Table 3-1d
VYNPS Upper Shelf Energy Equivalent Margin Analysis
40-Year Life (33 EFPY; 4.827E+8 MWH)

Equivalent Margin Analysis
Plant Applicability Verification Form
for Vermont Yankee
Including Extended Power Uprate Conditions
40-Year Life (33 EFPY; 4.827E+8 MWH)

BWR/2-6 WELD

Surveillance Weld USE (SMAW):

%Cu	=	0.03	
1st Capsule Fluence	=	4.49E+16 n/cm^2	
1st Capsule Measured % Decrease	=	4.80	(Charpy Curves)
1st Capsule RG 1.99 Predicted % Decrease	=	4.77	(RG 1.99, Rev. 2, Figure 2)
Ratio of Measured to Predicted % Decrease	=	1.005	(RG 1.99, Rev. 2, Position 2.2)

Limiting Beltline Weld USE (SMAW):

%Cu	=	0.04	
33 EFPY (4.827E+8 MWH) 1/4T Fluence	=	2.35E+17 n/cm^2	
RG 1.99 Predicted % Decrease	=	7.43	(RG 1.99, Rev. 2, Figure 2)
Adjusted % Decrease	=	7.47	(RG 1.99, Rev. 2, Position 2.2)

7.47% $\leq$ 34%

Therefore, vessel welds are bounded by Equivalent Margin Analysis

Table 3-1e
VYNPS Upper Shelf Energy Equivalent Margin Analysis Summary

Equivalent Margin Upper Shelf Energy Summary

NEDO-32205 Appendix B Worksheet					
Surveillance Information	Cu (%)	Capsule Fluence (n/cm^2)	Measured Decrease (%)	RG1.99 Predicted Decrease (%)	Ratio of Measured to Predicted (F1)
Plate	0.11	4.49E+16	8.03	5.55	1.448
Weld	0.03	4.49E+16	4.80	4.77	1.005
32 EFPY CLTP Beltline Material Information	Cu (%)	EOL 1/4T Fluence (n/cm^2)	RG1.99 Predicted Decrease (%)	Adjusted Decrease = Predicted * F1 (%)	NEDO-32205 Limit (%)
Plate	0.14	2.21E+17	9.35	13.50	21
Weld	0.04	2.21E+17	7.32	7.36	34
33 EFPY (4.827E+8 MWH) Beltline Material Information	Cu (%)	EOL 1/4T Fluence (n/cm^2)	RG1.99 Predicted Decrease (%)	Adjusted Decrease = Predicted * F1 (%)	NEDO-32205 Limit (%)
Plate	0.14	2.35E+17	9.50	13.80	21
Weld	0.04	2.35E+17	7.43	7.47	34

Table 3-2a
VYNPS Adjusted Reference Temperatures
40-Year Life (32 EFPY; CLTP)

Lower-Intermediate Shell Plates and Welds

Thickness in inches = 5.06

Ratio Peak/ Location = 1.00

32 EFPY Peak I.D. fluence = 2.99E+17 n/cm^2
32 EFPY Peak 1/4 T fluence = 2.21E+17 n/cm^2

Lower Shell Plates

Thickness in inches= 5.06

Ratio Peak/ Location = 0.74

32 EFPY Peak I.D. fluence = 2.21E+17 n/cm^2
32 EFPY Peak 1/4 T fluence = 1.63E+17 n/cm^2

COMPONENT	HEAT OR HEAT/LOT	%Cu	%Ni	CF	Initial RT_{NDT} °F	1/4 T Fluence n/cm^2	32 EFPY Δ RT_{NDT} °F	σ_I	σ_Δ	Margin °F	32 EFPY Shift °F	32 EFPY ART °F
PLATES:												
Lower Shell												
1-17	C2640	0.12	0.61	83	0	1.63E+17	12.57	0	6.29	12.57	25.14	25.14
1-16	C2653	0.13	0.59	91	0	1.63E+17	13.78	0	6.89	13.78	27.57	27.57
Lower-Intermediate Shell												
1-15	C3116	0.14	0.66	102	-10	2.21E+17	18.65	0	9.32	18.65	37.29	27.29
1-14	C3017	0.11	0.63	74	30	2.21E+17	13.61	0	6.80	13.61	27.22	57.22
WELDS:												
	SMAW	0.04	1.00	54	-70	2.21E+17	9.87	0	4.94	9.87	19.74	-50.26

Table 3-2b
VYNPS Adjusted Reference Temperatures
40-Year Life (33 EFPY; 4.827E+8 MWH)

Lower-Intermediate Shell Plates and Welds

Thickness in inches = 5.06 | Ratio Peak/ Location = 1.00 | 33 EFPY Peak I.D. fluence = 3.18E+17 n/cm^2
33 EFPY Peak 1/4 T fluence = 2.35E+17 n/cm^2

Lower Shell Plates

Thickness in inches= 5.06 | Ratio Peak/ Location = 0.74 | 33 EFPY Peak I.D. fluence = 2.35E+17 n/cm^2
33 EFPY Peak 1/4 T fluence = 1.74E+17 n/cm^2

COMPONENT	HEAT OR HEAT/LOT	%Cu	%Ni	CF	Initial RT_{NDT} °F	1/4 T Fluence n/cm^2	33 EFPY Δ RT_{NDT} °F	σ_I	σ_Δ	Margin °F	33 EFPY Shift °F	33 EFPY ART °F
PLATES:												
Lower Shell												
1-17	C2640	0.12	0.61	83	0	1.74E+17	13.07	0	6.54	13.07	26.15	26.15
1-16	C2653	0.13	0.59	91	0	1.74E+17	14.33	0	7.17	14.33	28.67	28.67
Lower-Intermediate Shell												
1-15	C3116	0.14	0.66	102	-10	2.35E+17	19.36	0	9.68	19.36	38.72	28.72
1-14	C3017	0.11	0.63	74	30	2.35E+17	14.13	0	7.06	14.13	28.26	58.26
WELDS:												
	SMAW	0.04	1.00	54	-70	2.35E+17	10.25	0	5.12	10.25	20.50	-49.50

Table 3-3
VYNPS CUFs of Limiting Components [1]

	P + Q Stress (ksi)			CUF		
Component	**CLTP**	**CPPU**	**Allowable (ASME Code Limit)**	**CLTP**	**CPPU**	**Allowable**
Feedwater Nozzle				0.50	0.75	1.0
Nozzle	66.2	80.0	80.1 ($3S_m$)			
Safe End	42.9	50.4	54.3 ($3S_m$)			
CS Nozzle				0.63	0.63	1.0
Nozzle	52.2	52.2	80.0 ($3S_m$)			
Safe End	65.2	65.2	70.0 ($3S_m$)			
Main Closure Stud	92.1	92.1	118.9 ($3S_m$)	0.62	0.62	1.0
Recirculation Inlet Nozzle				0.61	0.61	1.0
Safe End	50.4	50.4	51.75 ($3S_m$)			
Overlay [2]	48.4	48.4	49.8 ($3S_m$)			
Overlay [2]	51.5	51.5	80.1 ($3S_m$)			
Recirculation Outlet Nozzle				0.87	0.87	1.0
Nozzle	27.9	27.9	80.1 ($3S_m$)			
Safe End	46.7	46.7	51.75 ($3S_m$)			

Notes:

1. Only components with usage factors greater than [[ ]] are included in this table.
2. Weld overlay (internal) location at nozzle to safe end weld (part of Recirculation inlet safe end modification).

Table 3-4
VYNPS RIPDs for Normal Conditions (psid)

Parameter	CLTP	CPPU [1.]
Core Plate and Guide Tube	23.43	24.40
Shroud Support Ring and Lower Shroud	27.80	29.31
Upper Shroud	4.37	4.90
Shroud Head	4.56	5.40
Shroud Head to Water Level (Irreversible [2.])	6.53	7.69
Shroud Head to Water Level (Elevation [2.])	0.76	0.67
Top Guide	0.53	0.62
Steam Dryer	0.35	0.45
Fuel Channel Wall	12.24	13.32

Notes:

1. 107% core flow.
2. Irreversible loss is the loss across the separators; the elevation loss or reversible head loss is the loss between the inside shroud to the exit of the separators.

Table 3-5
VYNPS RIPDs for Upset Conditions (psid)

Parameter	CLTP	CPPU [1]
Core Plate and Guide Tube	25.83	26.80
Shroud Support Ring and Lower Shroud	30.20	31.71
Upper Shroud	6.56	7.36
Shroud Head	6.84	8.11
Shroud Head to Water Level (Irreversible [2])	9.79	11.53
Shroud Head to Water Level (Elevation [2])	1.14	1.01
Top Guide	1.14	0.71
Steam Dryer	0.53	0.59
Fuel Channel Wall	15.14	16.22

Notes:

1. 107% core flow.
2. Irreversible loss is the loss across the separators; the elevation loss or reversible head loss is the loss between the inside shroud to the exit of the separators.

Table 3-6
VYNPS RIPDs for Faulted Conditions (psid)

Parameter	CLTP	CPPU [1.]
Core Plate and Guide Tube	33.0	33.0
Shroud Support Ring and Lower Shroud	48.0	48.0
Upper Shroud	26.0	25.5
Shroud Head	25.0	25.0
Shroud Head to Water Level (Irreversible [2.])	26.5	26.0
Shroud Head to Water Level (Elevation [2.])	2.0	2.0
Top Guide	1.4	1.0
Steam Dryer	6.8	6.9
Fuel Channel Wall	16.8	17.0

Notes:

1. 107% core flow.

2. Irreversible loss is the loss across the separators; the elevation loss or reversible head loss is the loss between the inside shroud to the exit of the separators.

Table 3-7
VYNPS Reactor Internal Components - Summary of Stresses

Item	Component Location	Service Condition	Stress/Load Category	CLTP Basis Value	CPPU Value	Allowable For CPPU
1	Shroud / Shroud Repair	Normal Upset Faulted	Vertical Loads (lbs)	54.8 71.8 178.9	59.8 81.2 187.7	94.2 94.2 197.9
2	Shroud Support	Normal Upset Faulted	Vertical Loads (lbs)	54.8 71.8 178.9	59.8 81.2 187.7	102.4 102.4 272.2
3	Core Plate (including core plate plugs)	Normal /Upset	Buckling/Sliding (psid)	25.8/25.8	26.8/26.8	29.1/30.7 (Equivalent Allowable Pressure)
4	Core Plate (including core plate plugs)	Emergency	Buckling/Sliding (psid)	Qualified by Qualitative Assessment	33.0/24.4	38.9/45.9 (Equivalent Allowable Pressure)
5	Core Plate (including core plate plugs)	Faulted	Buckling/Sliding (psid)	33.0/33.0	33.0/33.0	51.9/43.8 (Equivalent Allowable Pressure)
6	Top Guide	Normal/Upset /Emergency/ Faulted	Bounded by CLTP design basis Loads/Stresses			
7	CRD Housing and CRD	Normal/Upset /Emergency/ Faulted	Loads (lbs.)	Qualified by Qualitative Assessment	Qualified By Qualitative Assessment (see Sections 2.5.3 and 3.3.2)	
8	Control Rod Guide Tube	Normal/Upset	Buckling	0.41	0.41	0.45
9	Control Rod Guide Tube	Emergency	Buckling	Qualified by Qualitative Assessment	0.41	0.60
10	Control Rod Guide Tube	Faulted	Buckling	0.67	0.67	0.80
11	Orificed Fuel Support	Normal/Upset	Stress (psi)	Qualified by Qualitative Assessment	7,275	15,580
12	Orificed Fuel Support	Emergency	Stress (psi)	Qualified By Qualitative Assessment (see Section 3.3.2)		
13	Orificed Fuel Support	Faulted	Stress (psi)	Qualified by Qualitative Assessment	15,349	35,440

Item	Component Location	Service Condition	Stress/Load Category	CLTP Basis Value	CPPU Value	Allowable For CPPU
14	Fuel Channel	Normal/Upset /Emergency/ Faulted	Qualified per Proprietary Fuel Design Basis			
15	Steam Dryer (Lifting Rod)	Normal/Upset	Buckling (lbs./Rod)	Qualified by Qualitative Assessment	6,041	35,300
16	Steam Dryer (Lifting Rod)	Emergency	Buckling (lbs./Rod)	Qualified by Qualitative Assessment	42,087	52,950
17	Steam Dryer (Lifting Rod)	Faulted	Buckling (lbs./Rod)	Qualified by Qualitative Assessment	55,010	70,600
18	Steam Dryer (Hood)	Normal/Upset /Emergency/ Faulted	Qualified by Qualitative Assessment for CLTP and CPPU (see Section 3.3.2)			
19	Feedwater Sparger	Normal/Upset Emergency Faulted	Qualified by Qualitative Assessment for CLTP and CPPU (see Section 3.3.2)			
20	Jet Pump (Beam Bolt Preload, Riser Elbow to Thermal Sleeve)	Normal/Upset /Emergency/ Faulted	Qualified by Qualitative Assessment for CLTP and CPPU (see Section 3.3.2)			
21	Jet Pump (Diffuser)	Normal/Upset /Emergency	Qualified by Qualitative Assessment for CLTP and CPPU (see Section 3.3.2)			
22	Jet Pump (Diffuser)	Faulted	P_b (psi)	Qualified by Qualitative Assessment	40,649	48,000
23	CS Line and Sparger	Qualified by Qualitative Assessment for CLTP and CPPU (see Section 3.3.2)				
24	Access Hole Cover	Normal/Upset	Stresses (psi)	Qualified by Qualitative Assessment	1,746	14,000
25	Access Hole Cover	Emergency/ Faulted	Stresses (psi)	Qualified by Qualitative Assessment	11,825	47,300
26	Shroud Head and Steam Separator Assembly (including SHBs)	Normal/Upset	Load (lbs.)	7,574	10,181	15,500 (Preload)

Item	Component Location	Service Condition	Stress/Load Category	CLTP Basis Value	CPPU Value	Allowable For CPPU
27	Shroud Head and Steam Separator Assembly (including SHBs)	Emergency	Pm (psi)	6,458	Bounded by CLTP	34,950
28	Shroud Head and Steam Separator Assembly (including SHBs)	Faulted	Pm (psi)	9,569	Bounded by CLTP	46,600
29	In-Core Housing and Guide Tube	Normal/Upset Emergency/ Faulted	Qualified by Qualitative Assessment for CLTP and CPPU (see Section 3.3.2)			

Table 3-8a
VYNPS BOP Piping
FW, Extraction Steam, FW Heater Drains and Vents, and Condensate

Piping System	Loading Condition	CPPU Stress (psi)	Allowable Stress (psi)	Design Margin [1.]
Feedwater	Thermal	20,243	22,500	0.90
Extraction Steam	Thermal	4,695	22,500	0.21
FW Heater Vents and Drains	Thermal	6,145	22,500	0.27
Condensate	Thermal	5,066	22,500	0.23

Note:

1. Design Margin = CPPU Stress/Allowable Stress.

Table 3-8b
VYNPS BOP Piping
Torus Attached Piping

Torus Penetration (Piping System)	Loading Condition	CPPU Stress (psi)	Allowable Stress (psi)	Design Margin [1]
X-202A-F (Primary Containment and Atmospheric Control (PCAC))	Thermal	28,315	37,500	0.76
X202H&K (PCAC)	Thermal	33,028	37,500	0.88
X-205 (PCAC)	Thermal	27,089	37,500	0.90
X-210A&211A (CS/RHR)	Thermal	32,701	37,500	0.87
X-210B&211B (CS/RHR)	Thermal	32,962	37,500	0.88
X-212 (RCIC)	Thermal	9,498	22,500	0.42
X-216 (Sampling)	Thermal	12,599	22,500	0.56
X-220 (Sampling)	Thermal	19,714	22,500	0.88
X-224A (RHR)	Thermal	29,637	37,500	0.79
X-224B (RHR)	Thermal	35,877	37,500	0.96
X-225 (HPCI)	Thermal	15,087	22,500	0.67
X-226A (CS)	Thermal	21,435	22,500	0.95
X-226B (CS)	Thermal	21,671	22,500	0.96
X-227 (RCIC)	Thermal	28,702	37,500	0.77
X-232 (RCIC)	Thermal	17,724	22,500	0.79
X-233 (HPCI)	Thermal	5,058	22,500	0.22

Note:

1. Design Margin = CPPU Stress/Allowable Stress.

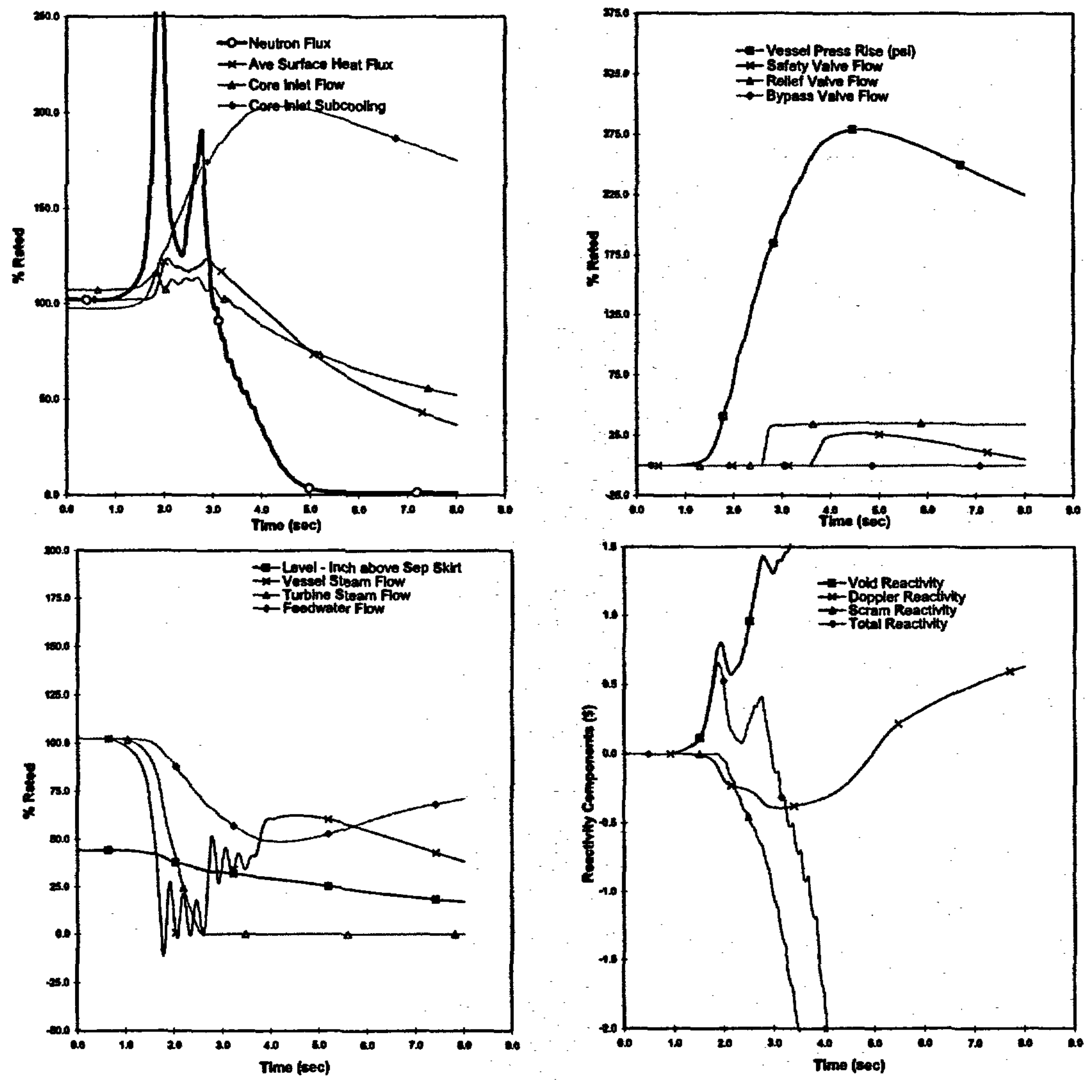
Table 3-8c
VYNPS BOP Piping
Main Steam System
(Outside Containment)

Loading Condition	**CPPU Stress (psi)**	**Allowable Stress (psi)**	**Design Margin [1.]**
Deadweight (DWT) + Thermal (TH) + TSV Closure + design basis earthquake	16,432	18,000	0.91
DWT + TH + TSV Closure + maximum hypothetical earthquake	18,042	30,000	0.60

Note:

1. Design Margin = CPPU Stress/Allowable Stress.

Figure 3-1
VYNPS Response to MSIV Closure with Flux Scram
(102% CPPU power, 107% core flow, and 1040 psia initial dome pressure)



4. ENGINEERED SAFETY FEATURES

This section primarily focuses on the information requested in RG 1.70, Chapter 6, which applies to CPPU. RG 1.70, Chapter 6 states, "engineered safety features are provided to mitigate the consequence of postulated accidents," and "are those (features) that are commonly used to limit the consequences of postulated accidents." NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 6.1.1, subsection I states, "Engineered safety features (ESF) are provided in nuclear plants to mitigate the consequences of design basis or loss-of-coolant accidents." The VYNPS plant features evaluated within this section are designed to (directly) mitigate the consequences of postulated accidents, and thus, are classified in the plant UFSAR as engineered safety features, consistent with RG 1.70 and NUREG-0800.

4.1 CONTAINMENT SYSTEM PERFORMANCE

This section addresses the effect of the CPPU on various aspects of the VYNPS containment system performance. The topics addressed in this evaluation are:

Topic		CPPU Disposition	VYNPS Result
4.1.1	Pool temperature response	[[	
4.1.1	Wetwell pressure		
4.1.1	Drywell temperature		
4.1.1	Drywell pressure		
4.1.2	Containment dynamic loads		
4.1.3	Containment isolation		
4.1.4	Motor-operated valves		
4.1.5	Hardened wetwell vent system		
4.1.6	Equipment operability		]]

The UFSAR provides the containment responses to various postulated accidents that validate the design basis for the containment. Operation at the CPPU RTP causes changes to some of the conditions for the containment analyses. For example, the short-term DBA LOCA containment response during the reactor blowdown is governed by the blowdown flow rate. This blowdown flow rate is dependent on the reactor initial thermal-hydraulic conditions, such as vessel dome pressure and the mass and energy of the vessel fluid inventory, which change slightly at the CPPU RTP. Also, the long-term heatup of the suppression pool following a LOCA or a transient is governed by the ability of the RHR system to remove decay heat. Because the decay heat depends on the initial reactor power level, the long-term containment response is affected by CPPU. The containment pressure and temperature responses have been reanalyzed, as described in Section 4.1.1, to demonstrate the VYNPS capability to operate at CPPU RTP.

The analyses were performed in accordance with RG 1.49 and References 2 and 3 using GE codes and models (References 13 through 16). The application of the GE methods to CPPU evaluations have been reviewed and approved by the NRC (References 3, 17, 18, and 19). The M3CPT code is used to analyze the short-term containment pressure and temperature response to the DBA-LOCA at CPPU conditions. This code was also used to analyze the short-term DBA-LOCA containment response for the UFSAR. The CPPU analysis used LAMB (Reference 16) with Moody's Slip critical flow model (Reference 15) to calculate the blowdown flow rates, which are then used as inputs to M3CPT. This approach, referred to as the "CPPU Method," differs from that for the current UFSAR analysis, which uses the Homogeneous Equilibrium Model (HEM). Application of the LAMB blowdown model for an EPU analysis is identified in ELTR1. The SHEX code was used for the CPPU long-term containment analysis. Confirmatory calculations with the SHEX code were performed at CLTP conditions to compare the SHEX result with the value of peak suppression pool temperature reported in the UFSAR. The comparison shows a difference of less than 0.5°F in peak suppression pool temperature. Therefore, the use of the SHEX code for VYNPS complies with the NRC requirements (Reference 17).

The effect of CPPU on the containment dynamic loads due to a LOCA or SRV discharge was also evaluated as described in Section 4.1.2. These loads were previously defined generically during the Mark I Containment Long Term Program (LTP) as described in Reference 20 and accepted by the NRC per References 21 and 22. Based on Reference 20, plant-specific dynamic loads for VYNPS were defined (Reference 23). The evaluation of the LOCA containment dynamic loads at CPPU conditions was based primarily on the results of the short-term pressure and temperature response analysis described in Section 4.1.1.3. The SRV discharge load evaluation is based on the CPPU condition of no changes in the SRV opening setpoints relative to the CLTP condition.

4.1.1 Containment Pressure and Temperature Response

Short-term and long-term containment analyses results are reported in the UFSAR. The short-term analysis is directed primarily at determining the drywell pressure response during the initial blowdown of the reactor vessel inventory to the containment following a large break inside the drywell. The long-term analysis is directed primarily at the suppression pool temperature response, considering the decay heat addition to the suppression pool. Peak values of the containment pressure and temperature responses to the DBA-LOCA are given in Table 4-1. The effect of CPPU on the events yielding the limiting containment pressure and temperature responses are discussed below.

4.1.1.1 Suppression Pool Temperature Response

(a) Bulk Pool Temperature

A long-term suppression pool bulk temperature response with CPPU was evaluated for the DBA LOCA. The analysis was performed at 102% of the CPPU RTP. The CPPU Method uses decay

heat values based on the ANSI/ANS 5.1-1979 decay heat model with a 2σ uncertainty added. As shown in Table 4-1, the peak suppression pool temperature was calculated to be 194.7°F at CPPU conditions. This peak temperature is below the suppression chamber design value of 281°F.

The DBA-LOCA was also analyzed for the CLTP, using the same method and input assumptions, to assess the CPPU effect on peak suppression pool temperature on a common analysis basis. The results for the CLTP and CPPU conditions are compared in Table 4-1. This comparison shows that the DBA-LOCA peak suppression pool temperature increases by 12.3°F due to the CPPU.

The DBA-LOCA containment responses were used in the evaluation of the available NPSH for the CS and the LPCI/RHR pumps. The analysis was performed using input assumptions such that the suppression pool temperature response is maximized, while minimizing the containment pressure response. The results of the NPSH evaluation are provided in Section 4.2.6.

(b) Local Pool Temperature with SRV Discharge

The local pool temperature limit for SRV discharge is specified in NUREG-0783, because of concerns over unstable condensation observed at high pool temperatures in plants without quenchers. Reference 24 provides a justification for the elimination of this limit for plants with quenchers on the SRV discharge lines. Supplement 1 of Reference 25 agreed with the Reference 24 assessment with respect to unstable condensation, but raised concerns over the possibility of steam ingestion into the ECCS suction if the quenchers are below the ECCS suction inlet. Because VYNPS has SRV quenchers, no evaluation of the local pool temperature limit is necessary to address the possibility for unstable condensation. However, it is necessary to ensure that steam ingestion in the ECCS suction line is not of concern during SRV steam discharge.

An evaluation of steam ingestion concerns was performed when the ECCS suction strainer was installed, and no adverse effect on the ECCS suction due to steam ingestion was predicted for VYNPS (Reference 26). Therefore, based on the results of this evaluation and Reference 24, the local pool temperature limit specified in NUREG-0783 can be eliminated.

4.1.1.2 Containment Airspace Temperature Response

The short-term DBA-LOCA containment response analysis, which covers the blowdown period during which the maximum drywell pressure and temperature occurs, shows that the peak DW airspace temperature exceeds the structural design value of 281°F, as shown in Table 4-1. However, the DW airspace temperature exceeds the shell design value of 281°F for less than 10 seconds, which is an insufficient duration to increase the DW shell temperature above the design value. The analysis was performed at 102% of CLTP RTP and 102% of the CPPU RTP conditions, using the method used during the Mark I Containment LTP, with the break flow and

enthalpy calculated using a more detailed RPV model (Reference 16). The results show that CPPU has a negligible effect on the peak drywell airspace temperatures during the DBA-LOCA.

For the drywell airspace temperature response, the most severe result was obtained from the analysis of small steam line breaks. Five steam line break sizes were analyzed: 0.02, 0.05, 0.1, 0.2 and 0.5 ft^2. The highest peak airspace temperature among the five break sizes was 337.1°F, which occurred prior to initiation of containment spray. The resultant peak shell temperature of 271.6°F is below the 281°F design value. The DW airspace temperature responses are used in the evaluation of the CPPU effect on Environmental Qualification (EQ). The results of the EQ evaluation are provided in Section 10.3.

The long-term wetwell airspace temperature essentially follows the suppression pool temperature, and its peak value for the DBA-LOCA is below the suppression chamber design temperature of 281°F, as shown in Table 4-1.

Thus, the containment airspace temperature responses during the DBA-LOCA and more severe steam line breaks are acceptable at CPPU conditions from the standpoint of the containment structural design temperature.

4.1.1.3 Short-Term Containment Pressure Response

A short-term containment response analysis was performed for the limiting DBA LOCA, which assumes a double-ended guillotine break of a recirculation suction line, to demonstrate that CPPU does not result in exceeding the containment design limits. The short-term DBA-LOCA analysis covers a blowdown period during which the peak drywell pressure and temperature occur. This analysis was performed at 102% of the CPPU RTP, using the method used during the Mark I Containment LTP, with the break flow and enthalpy calculated using a more detailed RPV model (Reference 16). The results of this short-term analysis are summarized in Table 4-1. The peak drywell pressure of 41.8 psig at CPPU conditions is below the design pressure of 56 psig. Also included in this table is the peak drywell pressure for the CLTP, as obtained with the CPPU Method. As the table shows, CPPU results in a 0.2-psi increase in the peak drywell pressure for the DBA-LOCA. Compared with the value reported in the UFSAR, the CPPU Method resulted in a 3.4-psi higher peak drywell pressure at CLTP. The difference in results indicates the higher degree of conservatism of the CPPU Method compared to the UFSAR method. This higher degree of conservatism is primarily attributed to the use of the Moody's slip critical flow model.

The wetwell airspace experiences a secondary long-term pressure peaking around the time at which the peak suppression pool temperature occurs. The value of this long-term peak wetwell pressure for the DBA-LOCA at CPPU conditions (13.9 psig) is 2.8 psi higher than the corresponding peak value at CLTP conditions, but is well below the design value of 56 psig.

Thus, the containment pressure response for the limiting DBA-LOCA is acceptable at CPPU conditions.

4.1.2 Containment Dynamic Loads

4.1.2.1 Loss-of-Coolant Accident Loads

The LOCA containment dynamic loads include pool swell, CO, and chugging (Reference 20). For a Mark I plant like VYNPS, the vent thrust loads are also evaluated, as discussed in Reference 20. Evaluation of the LOCA dynamic loads for CPPU is primarily based on the short-term DBA-LOCA pressure and temperature response analysis. This analysis is performed as described in Section 4.1.1.3, using the Mark I Containment LTP method, except that the break flow is calculated using a more detailed RPV model (Reference 16). The application of this model to CPPU containment evaluations is discussed in ELTR1. The DBA-LOCA pressure and temperature response analyses provide the calculated values of the controlling parameters for the dynamic loads throughout the blowdown. The key parameters are drywell and wetwell pressure, vent flow rates, and suppression pool temperature.

The short-term DBA-LOCA containment responses for CPPU are within the range of test conditions used to define the pool swell and CO loads for VYNPS. The containment responses with CPPU, in which chugging would occur, are within the conditions used to define the chugging loads. The vent thrust loads with CPPU are calculated to be less than plant-specific values defined for VYNPS (Reference 23) during the Mark I Containment LTP.

Therefore, the existing definitions for the LOCA dynamic loads remain applicable at CPPU conditions.

4.1.2.2 Safety Relief Valve Loads

The SRV air-clearing loads include SRVDL loads, suppression pool boundary pressure loads, and drag loads on submerged structures. These loads are affected by the SRV opening setpoint pressure, water leg length in the SRVDL at the time of SRV opening, and SRVDL and suppression pool geometry. The SRV loads were evaluated for two different actuation phases: initial actuation and re-actuation.

For the initial SRV actuation following an event involving RPV pressurization, the only parameter change potentially introduced by CPPU, which can affect the SRV loads, is an increase in the SRV opening setpoint pressure. However, CPPU does not include an increase in the SRV opening setpoint pressures. Therefore, the SRV loads due to initial actuation for the CPPU remain bounded by the existing load definition.

After the SRV closes, water re-floods the SRVDL, as the steam in the line is condensed and low pressure is created. As the low pressure is created, the vacuum breaker in the SRVDL opens and the water level goes down. The current load definition for SRV re-actuation used the maximum reflood height, which depends upon the vacuum breaker capacity and line geometry. Because these parameters are not affected by the CPPU, the existing load definition for SRV re-actuation also remains applicable at CPPU conditions.

Therefore, the existing definitions for the SRV discharge loads remain applicable at CPPU conditions.

4.1.2.3 Subcompartment Pressurization

The design capability of the sacrificial shield wall surrounding the reactor vessel is 134 psid. UFSAR Section 12.3.5.2.1 describes the pressure differential due to a single-ended rupture of the 28-inch recirculation line as 110 psid for CLTP assuming 100% of the blowdown energy is discharged into the annulus. Under CPPU conditions, the blowdown flow rate would increase slightly due to the slight increase in subcooling in the water initially in the recirculation loops. The effect of the increase in subcooling would be less than 3 psid on the resulting annulus pressure, therefore adequate margin to the structural design value still remains.

4.1.3 Containment Isolation

The system designs for containment isolation are not affected by CPPU. The capabilities of isolation actuation devices to perform during normal operations and under post-accident conditions have been determined to be acceptable. Therefore, the VYNPS containment isolation capabilities are not adversely affected by the CPPU.

4.1.4 Generic Letter 89-10 Program

Motor Operated Valves

The GL 89-10 Program MOVs were evaluated for the effects of CPPU, including the effects of pressure locking and thermal binding per GL 95-07. The evaluation reviewed MOV system calculation inputs for CPPU related changes to the current analysis. These inputs included process pressure, process temperature, flow rate, valve differential pressure, ambient temperature, and motor voltage.

These inputs were reviewed for all valve-operating modes (including normal operating and post accident conditions). Based on this review, there are no changes to the design functional requirements of the MOVs.

For some of the MOVs, the CPPU conditions are predicted to result in minor process fluid condition changes or to increase ambient room temperatures (< 25°F for HELB events and < 10°F for LOCA events) at the MOV locations. The affected valves will be analyzed through MOV program calculation updates to determine any required adjustments. The program document updates and any resulting changes in the current MOV settings will be implemented prior to CPPU operation. CPPU MOV evaluations will be reflected in the VYNPS GL 89-10 program documentation. Based on the results of the program input review and evaluations, it is expected that modifications to MOV valve and / or motor sizes will not be required.

Air Operated Valves

Similar to the MOVs, Air-Operated Valves (AOVs) were reviewed to identify valves potentially affected by CPPU conditions. The review determined that CPPU will have limited, if any, affect on AOV design basis functions. Evaluation of affected valves may identify AOV or AOV controller setting changes and/or modifications. Any required control setting changes or modifications identified will be accomplished prior to CPPU implementation.

4.1.5 Generic Letter 89-16

In response to GL 89-16, VYNPS installed a hardened wetwell vent system. The hardened vent is designed to mitigate loss of Decay Heat Removal (DHR) by providing sufficient wetwell venting capability to prevent further containment pressurization with the containment at its pressure limit. According to GL 89-16, the vent should be designed with sufficient capacity to accommodate decay heat input equivalent to 1% of CLTP. At VYNPS, the hardened vent is capable of accommodating 1.3% decay heat input based on CLTP, and can therefore accommodate a power uprate of as much as 30% of CLTP. At the CPPU RTP conditions, the existing hardened wetwell vent will exhaust a smaller percentage of RTP. Based on the as-built design, the hardened wetwell vent will exhaust approximately 1.08% RTP at 1912 MWt (CPPU RTP).

4.1.6 Generic Letter 96-06

The VYNPS response to GL 96-06 was accomplished in part using the limiting drywell temperature, pressure, and steam mass fraction time histories for CLTP conditions. The results of the containment analysis presented within this section are bounded by the CLTP conditions assumed for the analysis of affected in-containment piping. Therefore, the existing VYNPS response to GL 96-06 remains valid for CPPU.

4.2 EMERGENCY CORE COOLING SYSTEMS

Each ECCS is discussed in the following subsections. The effect on the functional capability of each system, due to CPPU is addressed. The assumption of constant pressure minimizes the effect of CPPU for ECCS evaluation. The topics addressed in this evaluation are:

Topic	CPPU Disposition	VYNPS Result
4.2.1 High Pressure Coolant Injection	[[	
4.2.2 High Pressure Core Spray		
4.2.3 Core Spray		
4.2.4 Low Pressure Coolant Injection		
4.2.5 Automatic Depressurization		
4.2.6 ECCS Net Positive Suction Head		]]

4.2.1 High Pressure Coolant Injection

The HPCI system is designed to pump water into the reactor vessel over a wide range of operating pressures. The primary purpose of the HPCI system is to maintain reactor vessel coolant inventory in the event of a small break LOCA that does not immediately depressurize the reactor vessel. In this event, the HPCI system maintains reactor water level and helps depressurize the reactor vessel. The adequacy of the HPCI system is demonstrated in Section 4.3.

[[ ]] For CPPU, there is no change to the maximum nominal reactor operating pressure and the SRV setpoints remain the same. [[

]] The NPSH required by the HPCI pump [[

]]

4.2.2 High Pressure Core Spray

The High Pressure Core Spray system is not applicable to VYNPS.

4.2.3 Core Spray or Low Pressure Core Spray

The Low Pressure Core Spray (LPCS) system is not applicable to VYNPS.

The CS system is automatically initiated in the event of a LOCA. When operating in conjunction with other ECCS, the CS system is required to provide adequate core cooling for all LOCA events. There is no change in the reactor pressures at which the CS is required.

The CS system sprays water into the reactor vessel after it is depressurized. The primary purpose of the CS system is to provide reactor vessel coolant inventory makeup for a large break LOCA and for any small break LOCA after the reactor vessel has depressurized. It also provides long-term core cooling in the event of a LOCA. The CS system meets all applicable safety criteria for the CPPU. The adequacy of the CS system performance is demonstrated by the margins discussed in Section 4.3.

The higher suppression pool temperature (194.7°F) and containment pressure during a postulated LOCA (Section 4.1) do not affect hardware capabilities of CS equipment, except for the CS pump seals.

The peak suppression pool temperature during a limiting LOCA remains below the CS pump seal design temperature of 210°F. However, this temperature exceeds the maximum operating temperature of 185°F analyzed for the pump seals. Either the pump seals will be re-qualified for the peak suppression pool temperature, or a modification will be completed to ensure seal operation prior to the CPPU implementation.

[[

]]

4.2.4 Low Pressure Coolant Injection

The LPCI mode of the RHR system is automatically initiated in the event of a LOCA. The primary purpose of the LPCI mode is to help maintain reactor vessel coolant inventory for a large break LOCA and for any small break LOCA after the reactor vessel has depressurized. The LPCI operating requirements are not affected by CPPU. The adequacy of this system is demonstrated by the margins discussed in Section 4.3.

[[

]]

4.2.5 Automatic Depressurization System

The ADS uses SRVs to reduce the reactor pressure following a small break LOCA when it is assumed that the high-pressure systems have failed. This allows the CS and LPCI to inject coolant into the reactor vessel. The adequacy of this system is demonstrated by the margins discussed in Section 4.3. [[

]]

4.2.6 ECCS Net Positive Suction Head

Following a LOCA, the RHR and CS pumps operate to provide the required core and containment cooling. Adequate NPSH is required during this period to assure the essential pump

operation. The NPSH for the ECCS pumps was evaluated for the limiting conditions following a DBA LOCA. The limiting NPSH conditions occur during long-term post-LOCA pump operation and depend on the total pump flow rates, debris loading on the suction strainers, and suppression pool temperature.

The NPSH for each pump was calculated based on the expected flow rates during the short-term and long-term ECCS pump operation. The pump flow rates for the short-term case are 7400 (14,200) gpm total RHR flow for single (two) pump operation and 4600 gpm total CS flow. The pump flow rates for the long-term case are 7400 gpm total RHR flow and 3500 gpm total CS flow. The debris loading on the suction strainers and the methodology used to calculate available ECCS NPSH for CPPU are the same as the pre-CPPU conditions. The containment response temperature and pressure profiles are CPPU specific. CPPU RTP operation increases the reactor decay heat, which increases the heat addition to the suppression pool following a LOCA. As a result, the suppression pool water temperature and containment pressure increase.

The assumptions used in the CPPU containment response analyses maximize the suppression pool temperature and minimize the containment pressure. These include operation of the RHR pumps for containment cooling in the containment spray mode after 10 minutes. The analyses then assume that the operators establish long-term containment cooling and control ECCS flow.

Short-term and long-term containment analyses were performed for the CPPU conditions (short-term from 0 to 600 seconds and long-term from 0 until the end of the event). The short-term containment analysis shows that the peak suppression pool temperature of 165.1°F occurs at 600 seconds after the LOCA event when the suppression pool pressure is 17.75 psia. The long-term containment analysis shows that the peak suppression pool temperature of 194.7°F occurs at 24,094 seconds after the LOCA event when the suppression pool pressure is 22.77 psia. The NPSH analyses conclude that containment overpressure is needed to meet long-term NPSH requirements. Table 4-2 shows the overpressure credit and Figure 4-6 shows the containment overpressure available, the required overpressure, the overpressure credit, and NPSH margins during the long-term DBA LOCA at CPPU conditions.

Based on the above, VYNPS is requesting approval of the "stepped" overpressure credit shown in Table 4-2 and Figure 4-6 to meet DBA LOCA long-term NPSH requirements.

RHR is required to operate during an ATWS event. CPPU RTP operation increases the reactor decay heat, which increases the heat addition to the suppression pool following this event (see Section 9.3.1). As a result, the peak suppression pool water temperature and peak containment pressure increase. The NPSH evaluation at these peak pool temperatures shows that adequate overpressure is available to satisfy NPSH requirements for these pumps during an ATWS event.

RHR is also required to operate during SBO and Appendix R fire events. CS is required to operate during an Appendix R fire event. CPPU RTP operation increases the reactor decay heat, which increases the heat addition to the suppression pool following these events (see Sections 6.7.1 and 9.3.2). As a result, the peak suppression pool water temperature and peak

containment pressure increase. The NPSH evaluation at these peak pool temperatures shows that adequate overpressure is available to satisfy NPSH requirements for these pumps.

The HPCI system primary function is to provide reactor inventory makeup water and assist in depressurizing the reactor during an intermediate or small break LOCA. HPCI system operation is also credited during ATWS, Appendix R, and SBO events. The available NPSH and required NPSH for the HPCI pump are not changed for CPPU, because the system configuration and the specified operational temperature limit for the process water do not change.

4.3 EMERGENCY CORE COOLING SYSTEM PERFORMANCE

The VYNPS ECCS is designed to provide protection against postulated LOCAs caused by ruptures in the primary system piping. The ECCS performance characteristics will not be changed for CPPU. ECCS-LOCA performance analyses demonstrate that the 10 CFR 50.46 requirements continue to be met at the CPPU RTP conditions. The VYNPS topics addressed in this evaluation are:

Topic	CPPU Disposition	VYNPS Result
Large break peak clad temperature – limiting case	[[	
Large break peak clad temperature - limiting event analysis		
Small break peak clad temperature – break spectrum		
Small break peak clad temperature – ADS capacity		
Local cladding oxidation		
Core wide metal water reaction		
Coolable geometry		
Long-term cooling		]]

[[

]] The break spectrum response is determined by the ECCS network design and is common to all BWRs. [[

]]

The LBPCT was determined based on the calculated Appendix K PCT at rated core flow with an adder to account for uncertainties. The CPPU GE14 LBPCT is 1960°F at CPPU RTP and rated core flow. This is 50°F greater than the LBPCT at the pre-CPPU conditions. The CPPU GE13 LBPCT is 1940°F at CPPU RTP and rated core flow. This is 40°F greater than the LBPCT at the pre-CPPU conditions (see Table 4-3). The LBPCT for GE14 and GE13 fuel are bounding for GE9 fuel. Although the PCT changes due to CPPU are greater than the typically seen 20°F, these changes are small compared to the margin to the 2200°F licensing limit that the bounding LBPCTs of 1960°F and 1940°F provide. In addition, the effect on the LBPCT adder is negligible considering the margin to the 2200°F licensing limit. The ECCS-LOCA results for VYNPS are in conformance with the error reporting requirements of 10 CFR 50.46 through notification number 2003-003.

In addition to the large break LOCA analysis, the small break LOCA response was reviewed in order to assure adequate ADS capacity. [[

]] there is sufficient ADS capacity at CPPU conditions. Also, the plant performance improvement of one SRV OOS remains valid with CPPU.

For SLO, a multiplier is applied to the Two-Loop Operation (TLO) Planar Linear Heat Generation Rate (PLHGR) and MAPLHGR limits. The SLO PLHGR and MAPLHGR multiplier for each fuel type is set at a value that assures the expected SLO PCT will remain below the TLO PCT. Because the PCTs have increased for CPPU, and the operating conditions for SLO are not changed with CPPU, the current value for the SLO PLHGR and MAPLHGR multipliers remain acceptable for CPPU.

ARTS limits are unaffected by CPPU. Also, the effect of ICF on PCT is negligible with CPPU. Thus the ARTS limits, as well as the ICF domain, remain valid with CPPU.

4.4 MAIN CONTROL ROOM ATMOSPHERE CONTROL SYSTEM

The VYNPS topic addressed in this evaluation is:

Topic	CPPU Disposition	VYNPS Result
Iodine intake	[[	]]

The Control Room Heating, Ventilation, and Air Conditioning (HVAC) system is designed to provide appropriate temperature conditions for personnel and equipment in the Control Room during any mode of operation or the most adverse emergency condition. CPPU adds negligible heat loading to this area, as the majority of the heat loads are from components and personnel already in the Control Room prior to CPPU.

The Control Room HVAC system is also designed to allow manual isolation of the Control Room, from within the Control Room, by placing the HVAC system into the recirculation mode.

VYNPS has separately submitted an LAR (Reference 27) describing the full implementation of the Alternative Source Term (AST) methodology that complies fully with RG 1.183. The AST evaluation calculated the Control Room dose for all DBAs with the assumption that the post-isolation air inleakage is equal to the pre-isolation value. The evaluation also considered new source-to-receptor pathways to comprehensively evaluate the Control Room dose. The conservative results demonstrate that the CPPU dose to the Control Room occupants will be less than the 30-day 5-rem Total Effective Dose Equivalent (TEDE) dose limit for the limiting DBA LOCA. Table 4-4 summarizes the Control Room doses from the AST analyses.

4.5 STANDBY GAS TREATMENT SYSTEM

The Standby Gas Treatment System (SGTS) is designed to maintain secondary containment at a negative pressure and to filter the exhaust air for removal of fission products potentially present during abnormal conditions. By limiting the release of airborne particulates and halogens, the SGTS limits off-site dose following a postulated DBA. The topics addressed in this evaluation are:

Topic	CPPU Disposition	VYNPS Result
Flow capacity	[[	
Iodine removal capability		]]

The design flow capacity of the SGTS was selected to maintain the secondary containment at the required negative pressure to minimize the potential for exfiltration of air from the reactor building. [[

]] The total (radioactive plus stable) post-LOCA iodine loading on the charcoal adsorbers increases proportionally with the increase in core iodine inventory, which is proportional to core thermal power (Section 9.2). Sufficient charcoal mass is present so that the post-LOCA iodine loading on the charcoal remains below the guidance provided by RG 1.52.

While decay heat from fission products accumulated within the system filters and charcoal adsorbers increases in proportion to the increase in thermal power, the cooling air flow required to maintain components below operating temperature limits is well below the cooling flow capability of the system.

In support of the above conclusions, [[ ]] analysis was performed and documented in the CLTR to evaluate the SGTS at facilities that have received approval under 10 CFR 50.67 to implement an AST. [[

	]]

The results of the AST evaluation, applicable to VYNPS, show that the maximum charcoal loading, based on only 50 pounds of charcoal per adsorber train, is approximately [[ ]] of total iodine per gram of charcoal. This is [[ ]] the 2.5 mg/gm maximum value in RG 1.52. The maximum component temperature is approximately [[ ]] with normal flow

conditions and [[ ]] under conditions of a failed fan with minimum cooling flow, which is well below the [[ ]] charcoal ignition temperature.

[[

]]

4.6 MAIN STEAM ISOLATION VALVE LEAKAGE CONTROL SYSTEM

VYNPS does not use a Main Steam Isolation Valve Leakage Control System (MSIV-LCS).

4.7 POST-LOCA COMBUSTIBLE GAS CONTROL SYSTEM

The Combustible Gas Control System is designed to maintain the post-LOCA concentration of oxygen or hydrogen in the containment atmosphere below the lower flammability limit. The topics addressed in this evaluation are:

Topic	CPPU Disposition	VYNPS Result
System initiation time	[[	
Recombiner operating temperature		
Nitrogen makeup		]]

As a result of CPPU, the post-LOCA production of hydrogen and oxygen by radiolysis increases proportionally with power level. This increase in radiolysis has an effect on the time available to start the system before reaching the RG 1.7 lower flammability limit of 5% oxygen by volume in the containment. Under CPPU conditions, the required start time for the VYNPS Containment Atmosphere Dilution (CAD) system is 37 hours, a sufficient time period to allow operators to respond during the postulated LOCA event. The integrated hydrogen production rates from radiolysis and metal-water reaction are shown in Figure 4-1. Uncontrolled and controlled hydrogen and oxygen concentrations in the drywell and wetwell are shown in Figures 4-2 and 4-3.

The addition rate of nitrogen required to maintain the containment below the 5% oxygen lower flammability limit remains within the delivery capability of the system. Analysis of the containment pressure buildup as a result of continuing CAD operation shows that the containment repressurization limit of 28 psig (i.e., 50% of design pressure) is reached 35 days post-LOCA. The integrated nitrogen volume requirement and pressure response of the drywell are shown in Figures 4-4 and 4-5.

Therefore, the VYNPS Combustible Gas Control System is capable of meeting its design basis function of controlling oxygen concentration below the 5% lower flammability limit and thereby ensuring containment integrity following a postulated DBA LOCA under CPPU conditions.

Table 4-1
VYNPS Containment Performance Results

Parameter	CLTP from UFSAR	CLTP with CPPU Method [1]	CPPU	Design Limit
Peak Drywell Pressure (psig)	38.2	41.6	41.8	56
Peak Drywell Airspace Temperature (°F)	284	287.7	287.8	281 [2]
Peak Bulk Pool Temperature (°F)	182.6	182.4	194.7	281
Long-term Peak Wetwell Pressure [3] (psig)	N/A	11.1	13.9	56
Peak Wetwell Airspace Temperature (°F)	N/A	182.4	194.7	281

Notes:

1. The CPPU Method, which was used for the CPPU analysis, was used for CLTP to compare with the CPPU results on a common analysis basis.
2. The 281°F value is the structure design temperature. The DW airspace temperature exceeds the shell design value of 281°F for less than 10 seconds, which is an insufficient duration to increase the DW shell temperature above the design value.
3. The wetwell pressure peaks early in the event, and then peaks again around the time at which the wetwell airspace temperature peaks. This secondary peak temperature is presented as the long-term peak wetwell pressure for evaluation of the effect of CPPU.

Table 4-2
VYNPS Overpressure Credit for NPSH DBA LOCA – Long-Term

Time After LOCA (sec)	Overpressure Credit (psig)
601	2.4
2,000	2.4
2,001	3.4
4,000	3.4
4,001	4.4
6,000	4.4
6,001	5.1
9,000	5.1
9,001	6.1
40,000	6.1
40,001	5.6
50,000	5.6
50,001	5.1
60,000	5.1
60,001	4.6
70,000	4.6
70,001	4.1
80,000	4.1
80,001	3.6
90,000	3.6
90,001	3.1
110,000	3.1
110,001	2.6
130,000	2.6
130,001	2.1
150,000	2.1
150,001	1.7

Time After LOCA (sec)	Overpressure Credit (psig)
170,000	1.7
170,001	1.3
180,000	1.3

Table 4-3
VYNPS ECCS Performance Results

Parameter	CLTP	CPPU	10 CFR 50.46 Limit
Method	SAFER/GESTR	SAFER/GESTR	
Power	104.5% of CLTP	120% of CLTP	
1. Licensing Basis PCT (°F)	1900 (GE13) [1.] 1910 (GE14) [1.]	1940 (GE13) 1960 (GE14)	≤ 2200
2. Cladding Oxidation (% Original Clad Thickness)	< 3.0	< 3.0	≤ 17
3. Hydrogen Generation, Core-wide Metal-Water Reaction (%)	< 0.1	< 0.1	≤ 1.0
4. Coolable Geometry	OK	OK	Meet Criteria 1 and 2.
5. Core Long Term Cooling	OK	OK	Core flooded to TAF or Core flooded to jet pump suction elevation and at least one CS system is operating at rated flow.

Note:

1. The current power LBPCT at the 104.5% CLTP power level and rated flow conditions is calculated for the CPPU analysis. This allows for comparison with the CPPU LBPCT results, because existing current power LBPCT results are based on a different flow condition.

Table 4-4
VYNPS Control Room Dose from DBAs

DBA	Dose (rem TEDE)
LOCA (30 day)	3.4
MSLB [1.] (2 hour)	2.0
Control Rod Drop Accident (CRDA) (30 day)	0.4
Fuel Handling Accident (FHA) (2 hour)	0.153

Note:

1. MSLB results based on 4.0 μCi/gm I-131 Dose Equivalent.

Figure 4-1
VYNPS Time-integrated Containment Hydrogen Generation at CPPU

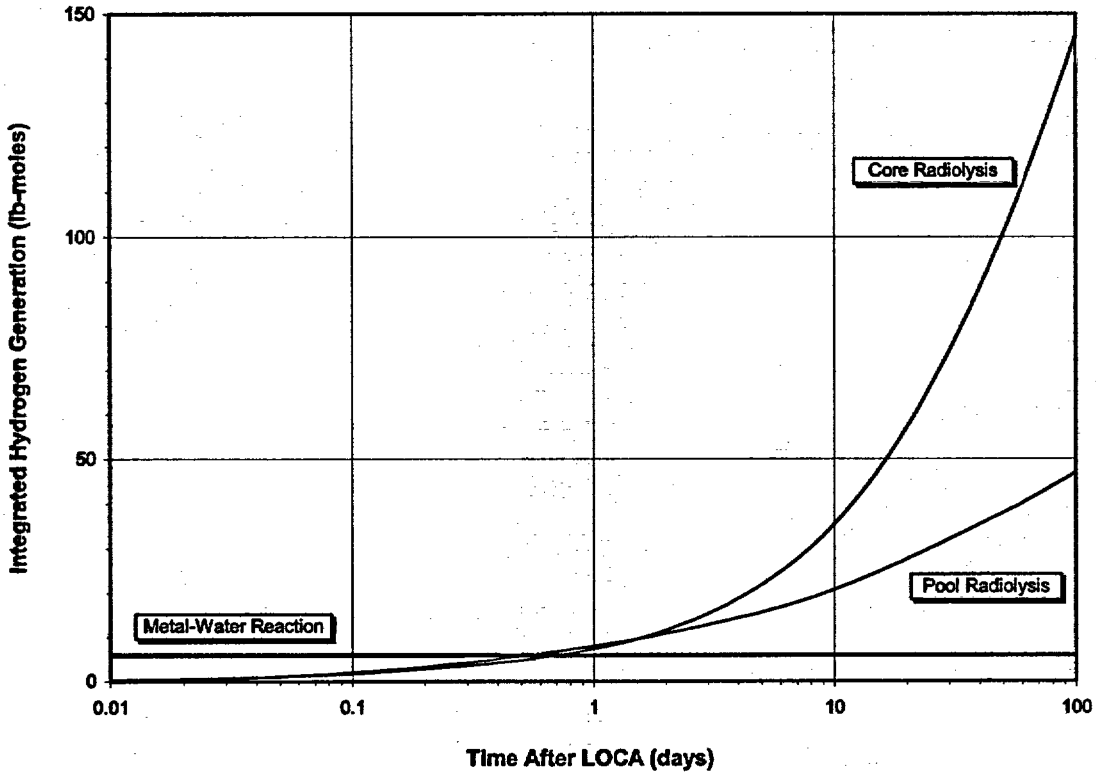


Figure 4-2
VYNPS Uncontrolled H_2 and O_2 Concentrations in Drywell and Wetwell at CPPU

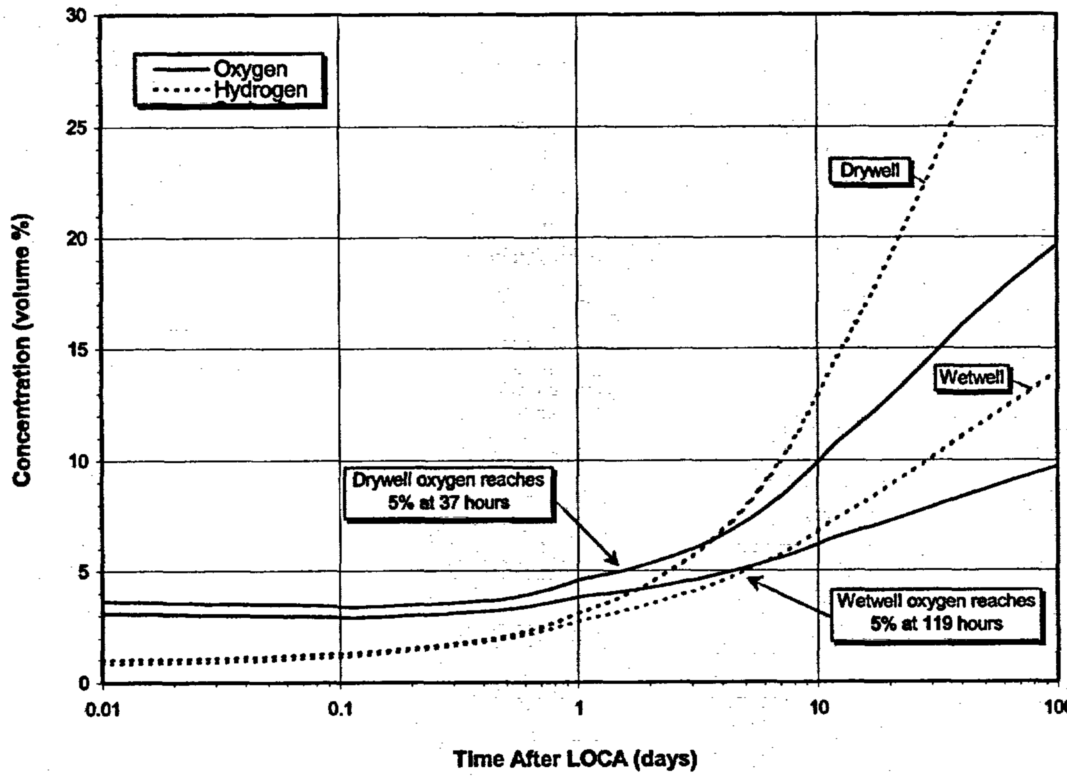


Figure 4-3
VYNPS Controlled H_2 and O_2 Concentrations in Drywell and Wetwell at CPPU

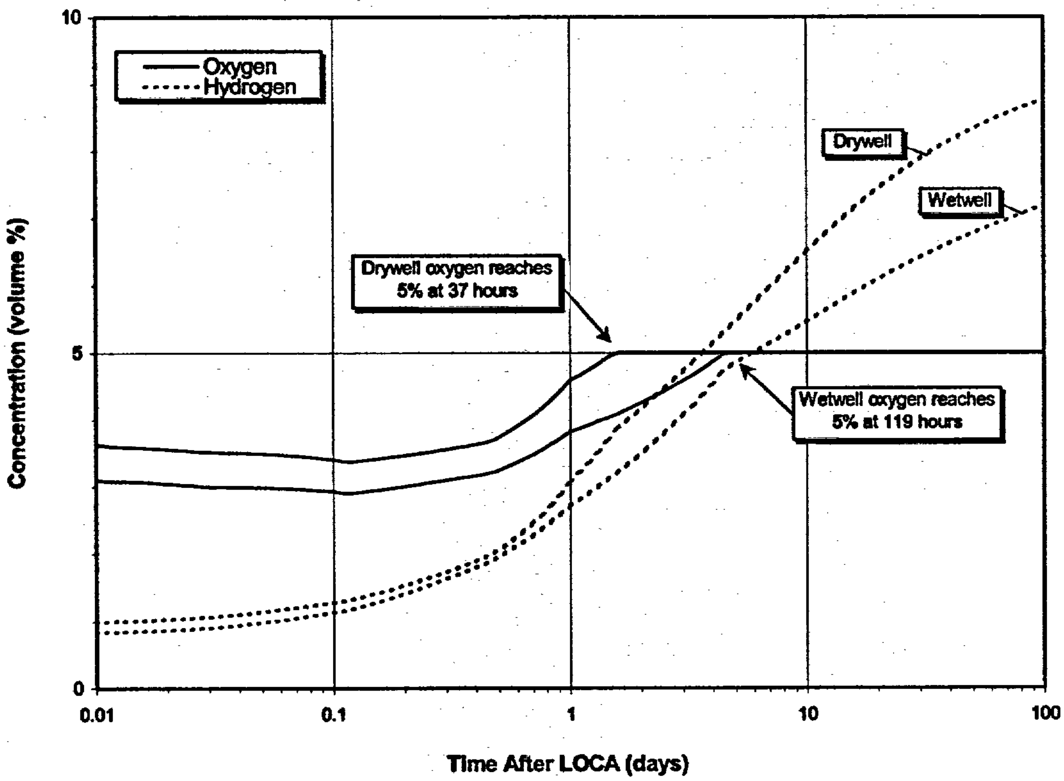


Figure 4-4
VYNPS CAD System Nitrogen Volume Requirement at CPPU

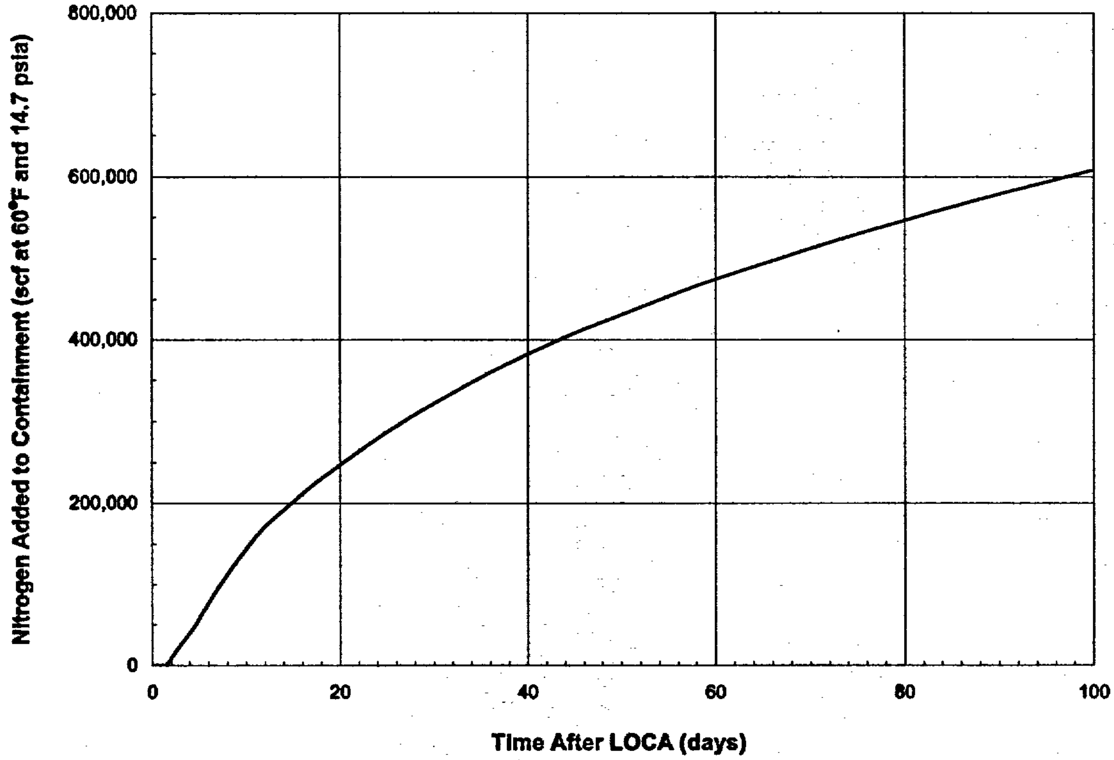


Figure 4-5
VYNPS Drywell Pressure Response to CAD Operation at CPPU

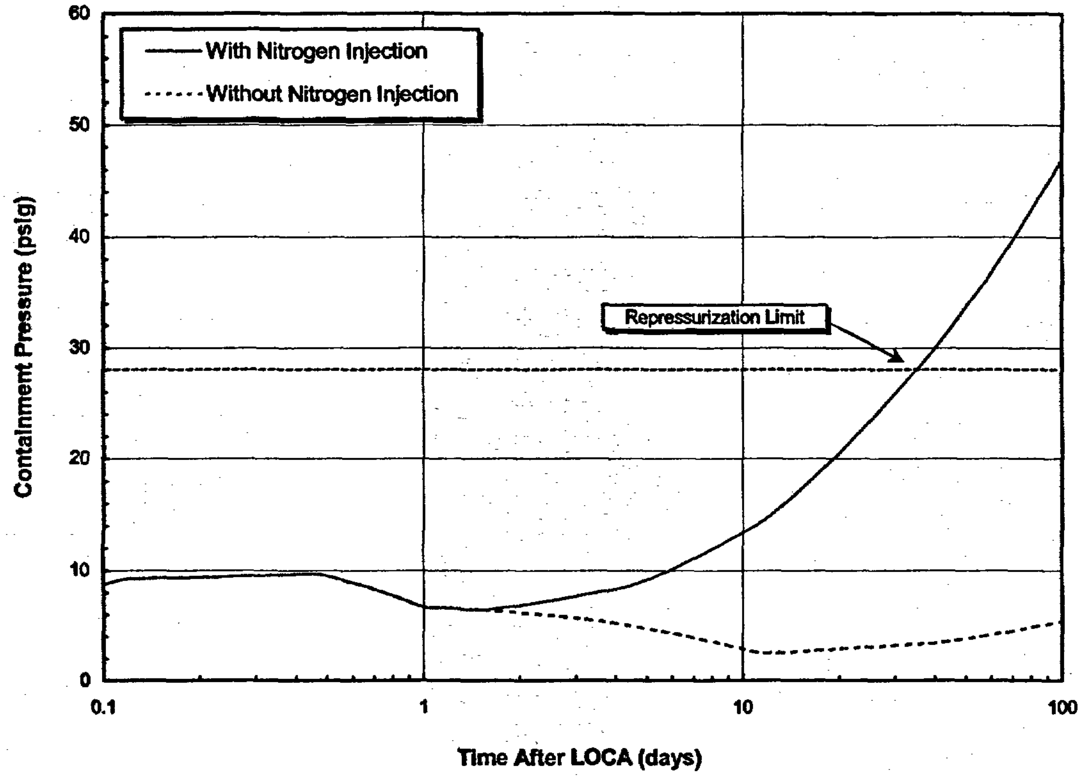
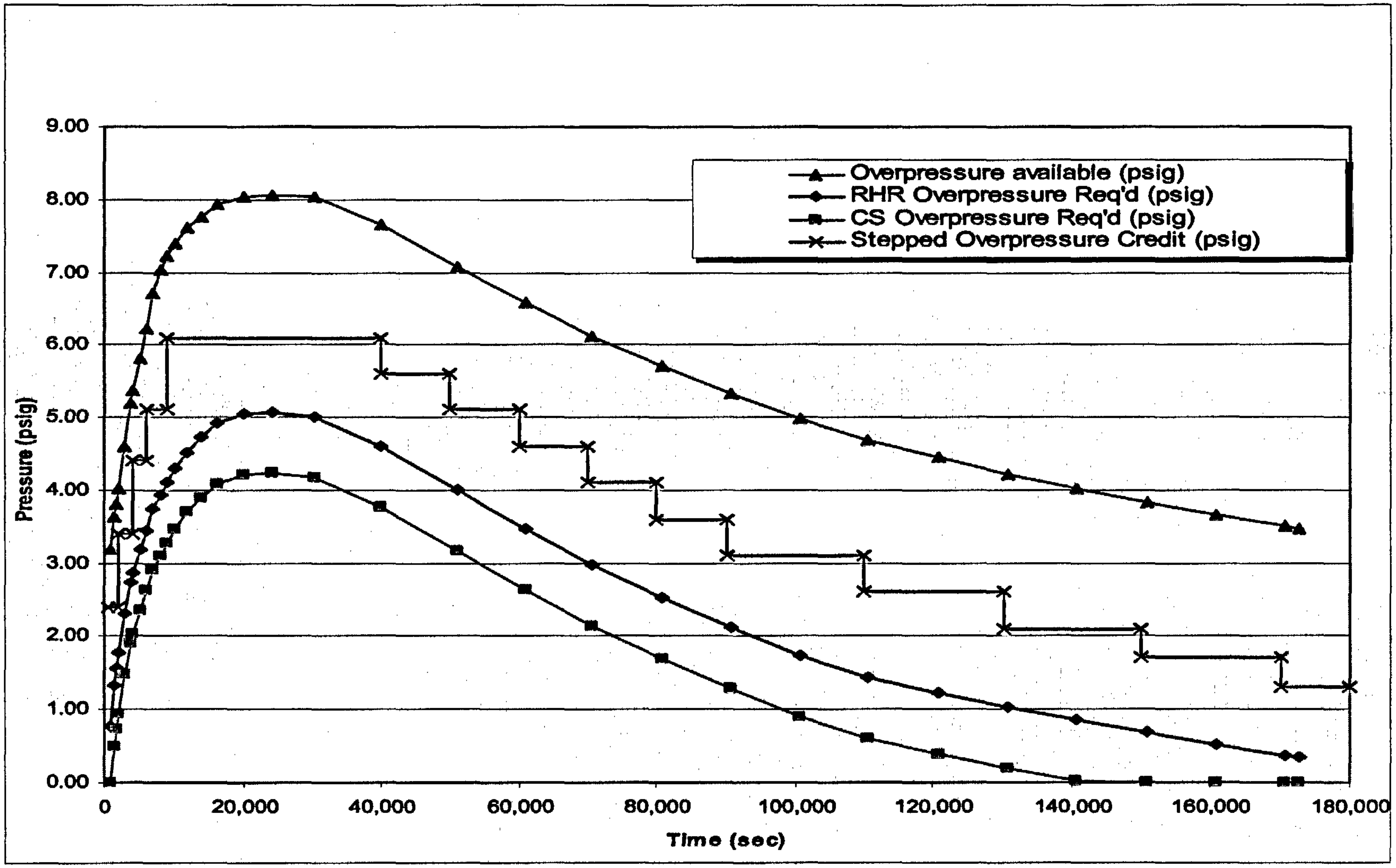


Figure 4-6
Overpressure Required for NPSH DBA LOCA – Long-Term



5. INSTRUMENTATION AND CONTROL

This section primarily focuses on the information requested in RG 1.70, Chapter 7, as it applies to CPPU. The principal instrumentation affected by CPPU is addressed in the following.

5.1 NSSS MONITORING AND CONTROL

The instruments and controls used to monitor and directly interact with or control reactor parameters are usually within the NSSS. Changes in process variables and their effects on instrument performance and setpoints were evaluated for CPPU operation to determine any related changes. Process variable changes are implemented through changes in normal plant operating procedures. Technical Specifications address instrument limits for those NSSS sensed variables that initiate protective actions. The effect of CPPU on Technical Specifications is addressed in Section 5.3. The topics addressed in this evaluation are:

Topic		CPPU Disposition	VYNPS Result
5.1.1.1	Average Power Range Monitors, Intermediate Range Monitors, and Source Range Monitors	[[	
5.1.1.2	Local Power Range Monitors		
5.1.1.3	Rod Block Monitor		
5.1.2	Rod Worth Minimizer/Rod Control Information System		]]

5.1.1 Neutron Monitoring System

CPPU affects the performance of the Neutron Monitoring System. These performance effects are associated with the APRMs, Intermediate Range Monitors (IRMs), Source Range Monitors (SRMs), Local Power Range Monitors (LPRMs), RBM, and Rod Worth Minimizer (RWM).

5.1.1.1 Average Power Range Monitors, Intermediate Range Monitors, and Source Range Monitors

The increase in power level due to CPPU increases the average flux in the core and at the in-core detectors. The APRM power signals are calibrated to read 100% at the new licensed power (i.e., CPPU RTP). CPPU has little effect on the IRM overlap with the SRMs and the APRMs. Using normal plant surveillance procedures, the IRMs may be adjusted, as required, so that overlap with the SRMs and APRMs remains adequate.

The SRM, IRM, and APRM Systems installed at VYNPS are in accordance with the requirements established by the GE design specifications. [[

]]

5.1.1.2 Local Power Range Monitors

At CPPU RTP, the average flux experienced by the detectors increases due to the average power increase in the core. The maximum flux experienced by an LPRM remains approximately the same because the peak bundle powers does not appreciably increase. Due to the increase in neutron flux experienced by the LPRMs and Traversing Incore Probes (TIPs), the neutronic life of the LPRM detectors may be reduced and radiation levels of the TIPs may be increased. LPRMs are designed as replaceable components. The LPRM accuracy at the increased flux is within specified limits, and LPRM lifetime is an operational consideration that is handled by routine replacement. TIPs are stored in shielded rooms. A small increase in radiation levels is accommodated by the radiation protection program for normal plant operation.

The LPRMs and TIPs installed at VYNPS are in accordance with the requirements established by the GE design specifications. [[

]]

5.1.1.3 Rod Block Monitor

The increase in power level at the same APRM reference level results in increased flux at the LPRMs that are used as inputs to the RBM. The RBM instrumentation is referenced to an APRM channel. Because the APRM has been rescaled, there is only a small effect on the RBM performance due to the LPRM performance at the higher average local flux. The change in performance does not have a significant effect on the overall RBM performance.

The RBMs installed at VYNPS are in accordance with the requirements established by the GE design specifications. [[

]] In addition, the RBM is not credited in any safety analysis for the VYNPS CPPU.

5.1.2 Rod Worth Minimizer/Rod Control and Information System

The Rod Control and Information System (RCIS) is not applicable to VYNPS.

The RWM is a normal operating system that does not perform a safety-related function. The function of the RWM is to support the operator by enforcing rod patterns until reactor power has reached appropriate levels. [[

]] The power-dependent instrument setpoints for the RWM are included in the plant Technical Specifications (see Section 5.3.4).

[[

]]

5.2 BOP MONITORING AND CONTROL

Operation of the plant at CPPU has minimal effect on the BOP system instrumentation and control devices. Based on CPPU operating conditions for the power conversion and auxiliary systems, most process control valves and instrumentation have sufficient range/adjustment capability for use at the CPPU conditions. However, some (non-safety) modifications may be needed to the power conversion systems to obtain CPPU RTP. No safety-related BOP system setpoint change is required as a result of the CPPU, with the exception of the main steam line high flow discussed in Section 5.3.1. The topics considered in this section are:

Topic	CPPU Disposition	VYNPS Result
5.2.1 Pressure Control System	[[	
5.2.2 Turbine Steam Bypass System (Normal Operation)		
5.2.2 Turbine Steam Bypass System (Safety Analysis)		
5.2.3 Feedwater Control System (Normal Operation)		
5.2.3 Feedwater Control System (Safety Analysis)		
5.2.4 Leak Detection System		]]

5.2.1 Pressure Control System

The Pressure Control System (PCS) is a normal operating system to provide fast and stable responses to system disturbances related to steam pressure and flow changes to control reactor pressure within its normal operating range. This system does not perform a safety function. Pressure control operational testing is included in the CPPU implementation plan as described in Section 10.4 to ensure adequate turbine control valve pressure control and flow margin is available.

[[

]]

5.2.2 Turbine Steam Bypass System

The Turbine Steam Bypass System is a normal operating system that is used to bypass excessive steam flow. The absolute flow capacity of the bypass system is unchanged. The bypass flow capacity is included in some AOO evaluations (Section 9.1). These evaluations demonstrate the adequacy of the bypass system. If the limiting event in the reload analysis takes credit for the availability of the bypass system, the bypass flow is used in the reload analysis to establish the core operating limits.

[[

]]

5.2.3 Feedwater Control System

The Feedwater Control System is a normally operating system to control and maintain the reactor vessel water level. CPPU results in an increase in feedwater flow. Feedwater control operational testing is included in the CPPU implementation plan as described in Section 10.4 to ensure that the feedwater response is acceptable. Failure of this system is evaluated in the reload analysis for each reload core with the feedwater controller failure-maximum demand event. An LOFW event can be caused by downscale failure of the controls. The LOFW is discussed in Section 9.1.3.

[[

]]

5.2.4 Leak Detection System

The only effect on the Leakage Detection System (LDS) due to CPPU is a slight increase in the feedwater temperature and steam flow. [[

]] The increased feedwater temperature results in a small increase in the main steam tunnel temperature (< 1.0°F). [[

]] Main steam line high flow is discussed in Section 5.3.1.

[[

]]

5.3 TECHNICAL SPECIFICATION INSTRUMENT SETPOINTS

Technical Specification instrument limits (TSLs) and/or setpoints are those sensed variables, which initiate protective actions and are generally associated with the safety analysis. TSLs are highly dependent on the results of the safety analysis. The safety analysis generally establishes the analytical limits (ALs). The determination of the instrument setpoints includes consideration

of measurement uncertainties and is derived from the ALs and TSLs. The settings are selected with sufficient margin to minimize inadvertent initiation of the protective action, while assuring that adequate operating margin is maintained between normal operating conditions and the Nominal Trip Setpoints (NTSPs).

Increases in the core thermal power and steam flow affect some instrument setpoints. These setpoints are adjusted to maintain comparable differences between the NTSP and TSLs, and are reviewed to ensure that adequate operational flexibility and necessary safety functions are maintained at the CPPU RTP level. When the power increase results in new instruments being employed, an appropriate setpoint calculation is performed and Technical Specification changes are implemented, as required. VYNPS has applied their existing setpoint methodology to the Technical Specification instrument setpoints affected by CPPU. All Technical Specification instruments were evaluated for the effects of CPPU. This evaluation included a review of environmental effects (i.e., radiation and temperature), process effects (i.e., measured parameter), and analytical effects (i.e., AL and margins) on the subject instruments.

Table 5-1 summarizes the current and CPPU ALs for VYNPS.

The topics addressed in this evaluation are:

Topic	CPPU Disposition	VYNPS Result
5.3.1 Main Steam Line High Flow Isolation - Setpoint Calculation Methodology	[[	
5.3.1 Main Steam Line High Flow Isolation - Setpoint Value		
5.3.2 Turbine First-Stage Pressure Scram Bypass - Setpoint Calculation Methodology		
5.3.2 Turbine First-Stage Pressure Scram Bypass - Setpoint Value		
5.3.3 APRM Flow-Biased Scram - Setpoint Calculation Methodology		
5.3.3 APRM Flow-Biased Scram - Setpoint Value		
5.3.4 Rod Worth Minimizer - Setpoint Calculation Methodology		
5.3.4 Rod Worth Minimizer - Setpoint Value		
5.3.5 Rod Block Monitor		

Topic	CPPU Disposition	VYNPS Result
5.3.6 RCIS Rod Withdrawal Limiter High Power Setpoint- Setpoint Calculation Methodology		
5.3.6 RCIS Rod Withdrawal Limiter High Power Setpoint- Setpoint Value		
5.3.7 APRM Setdown in Startup Mode - Setpoint Calculation Methodology		
5.3.7 APRM Setdown in Startup Mode - Setpoint Value		]]

Note:

1. [[

]]

5.3.1 Main Steam Line High Flow Isolation

5.3.1.1 In RUN Mode

The main steam line high flow isolation setpoint (in RUN mode) is used to initiate the isolation of the Group 1 primary containment isolation valves. The only safety analysis event that credits this trip is the MSLB accident. For this accident, there are diverse trips from high area temperature and high area differential temperature. For VYNPS, there is sufficient margin to choke flow, so the AL for CPPU is maintained at the current percent of rated steam flow in each main steam line.

For VYNPS, the AL of 143% and TSL of 140% of steam flow is not changed and no new instrumentation is required (the existing instrumentation has the required upper range limit to re-span the instrument loops to accommodate the new setpoint). A new setpoint will be calculated using the VYNPS setpoint methodology as noted in Section 5.3.

5.3.1.2 Not in RUN Mode

The main steam line high flow isolation setpoint (not in RUN mode) is used to initiate the isolation of the Group 1 primary containment isolation valves. The basis for the CLTP AL is to avoid an excessive depressurization and cooldown of the RPV in case of a pressure regulator failure (open) with the reactor mode switch not in the RUN position. For CPPU, the AL, for this trip is unchanged with respect to the absolute mass flow rate. The AL at CPPU (in percent steam flow) is therefore effectively reduced by the ratio of (CLTP steam flow) / (CPPU steam flow).

For VYNPS, the AL of 50% of CLTP steam flow remains the same in terms of absolute mass flow as the current AL, and is rescaled to 40.6% of CLTP steam flow. In addition, the TSL of 40% of steam flow is not changed. However, new instrumentation is required because the existing instrumentation has too large of a span to accommodate the new setpoint (and increased uncertainty). Therefore, a new setpoint will be calculated using the VYNPS setpoint methodology as noted in Section 5.3.

5.3.2 Turbine First-Stage Pressure Scram Bypass

CPPU results in an increased power level and the high-pressure turbine (HPT) modifications result in a change to the relationship of turbine first-stage pressure to reactor power level. The turbine first-stage pressure setpoint is used to reduce scrams at low power levels where the turbine steam bypass system is effective for TTs and generator load rejections. In the safety analysis, this trip bypass only applies to events at low power levels that result in a TT or load rejection. [[

]] Therefore, the TSL associated with this function becomes 25% of RTP. [[

]] A new setpoint will be calculated using the VYNPS setpoint methodology as noted in Section 5.3. The Technical Specification applicable condition in percent RTP was changed.

To assure that the new value is appropriate, CPPU plant ascension startup test or normal plant surveillance will be used to validate that the actual plant interlock is cleared consistent with the safety analysis.

5.3.3 APRM Flow Biased Scram

The APRM flow biased scram covers the low flow region of the power/flow map where the AL is based on stability requirements, and the high flow region where the AL is based on MELLLA requirements. The AL in the low flow region is defined by two lines as a function of drive flow, each with a different slope and intercept. In the high flow region, the AL is defined by a single line with its own slope and intercept. For all three lines, [[

]] The TSL remains equal to the AL.

A new setpoint will be calculated using the VYNPS setpoint methodology as noted in Section 5.3. The appropriate Technical Specification changes were made.

5.3.4 Rod Worth Minimizer/RCIS Rod Pattern Controller Low Power Setpoint

The RCIS Rod Pattern Controller is not applicable to VYNPS.

The RWM LPSP is used to bypass the rod pattern constraints established for the CRDA at greater than a pre-established low power level. The measurement parameters are feedwater and steam flow.

The LPSP AL and TSL in the plant Technical Specifications is [[

]] The TSL becomes 17% CPPU RTP. This approach does not affect the limitations on the sequence of control rod movement to the absolute core power level for the LPSP associated with the requirements of the CRDA. A new setpoint will be calculated using the VYNPS setpoint methodology as noted in Section 5.3.

5.3.5 Rod Block Monitor

The RBM rod block is no longer credited in the evaluation of the control rod withdrawal error as described in Section 3.4 of Reference 6. This was established in Reference 6 to ensure applicability of the off-rated thermal limits and the flexibility to relax the RBM setpoints. [[

]]

5.3.6 RCIS Rod Withdrawal Limiter High Power Setpoint

The RCIS Rod Withdrawal Limiter High Power Setpoint is not applicable to VYNPS.

5.3.7 APRM Setdown in Startup Mode

The value for the Technical Specification safety limit for reduced pressure or low core flow condition, is established to satisfy the fuel thermal limits monitoring requirements described in Section 2.1 of this document.

[[

]]

Table 5-1
VYNPS Analytical Limits For Setpoints

Parameter	Analytical Limits: Current	Analytical Limits: CPPU
APRM Calibration Basis	1593 MWt	1912 MWt
APRM High Flux Flow Bias (Scram) ALs		
TLO (%RTP)	$0.4\ W + 64.4\%$ for $0\% < W \leq 31.1\%$ $1.28\ W + 37.0\%$ for $31.1\% < W \leq 54.0\%$ $0.66\ W + 70.5\%$ for $54.0\% < W \leq 75.0\%$ Clamp at 120% for $W > 75.0\%$	$0.33\ W + 53.7\%$ for $0\% < W \leq 30.9\%$ $1.07\ W + 30.8\%$ for $30.9\% < W \leq 66.7\%$ $0.55\ W + 65.5\%$ for $66.7\% < W \leq 99.0\%$ Clamp at 120% for $W > 99.0\%$ [1.]
SLO (%RTP)	$0.4\ W + 61.2\%$ for $0\% < W \leq 39.1\%$ $1.28\ W + 26.8\%$ for $39.1\% < W \leq 61.9\%$ $0.66\ W + 65.2\%$ for $61.9\% < W \leq 83.0\%$ Clamp at 120% for $W > 83.0\%$	$0.33\ W + 51.1\%$ for $0\% < W \leq 39.1\%$ $1.07\ W + 22.2\%$ for $39.1\% < W \leq 61.7\%$ $0.55\ W + 54.3\%$ for $61.7\% < W \leq 119.4\%$ Clamp at 120% for $W > 119.4\%$ [1.]
APRM Setdown in Startup Mode (%RTP)		
Scram	15	No Change
Rod Block Monitor		
Auto Bypass (%RTP)	30	No Change
TLO and SLO Flow Biased (%RTP)	$0.66\ W + N\%$ [2.]	No Change [3.]
RBM Downscale	2 / 125 DIV	No Change [3.]
Rod Worth Minimizer LPSP (%RTP)	20	17
Main Steam Line High Flow Isolation (in RUN mode) (% rated steam flow)	143	No Change
Main Steam Line High Flow Isolation (not in RUN mode) (% rated steam flow)	50.0	40.6
Turbine First-Stage Pressure Scram Bypass (%RTP)	30%	25%

Notes:

[[

]]

2. "N" is specified in the COLR.
3. No credit is taken in any safety analysis for the RBM setpoints.

6. ELECTRICAL POWER AND AUXILIARY SYSTEMS

This section primarily focuses on the information requested in RG 1.70, Chapters 8 and 9, that applies to CPPU.

6.1 AC POWER

The VYNPS Alternating Current (AC) power supply includes both off-site and on-site power. The on-site power distribution system consists of transformers, buses, and switchgear. AC power to the distribution system is provided from the transmission system or from onsite Diesel Generators. Plant electrical characteristics are given in Table 6-1. The topics addressed in this evaluation are:

Topic	CPPU Disposition	VYNPS Result
AC power (degraded voltage)	[[	
AC power (normal operation)		]]

6.1.1 AC Power (degraded voltage)

The existing off-site electrical equipment was determined to be adequate for operation with the uprated electrical output as shown in Table 6-2. The review concluded the following.

The Isolated Phase Bus (IPB) duct is adequate for full generator output at all generator voltages. However, the IPB duct cooling system will be modified to increase its capacity to support CPPU operation.

The main generator step-up transformer has recently been replaced and was sized to support the CPPU operation. The associated switchyard components (rated for maximum transformer output) are adequate for the transformer output.

The main generator is being rewound and modified to support CPPU operation. Additionally, the bushing Current Transformers (CTs) will be replaced to support the new rewound main generator rating at full CPPU output. No generator protective relay changes are necessary, however some protective relaying set points will be modified for the rewound generator rating.

A grid stability analysis is being performed, considering the increase in electrical output, to demonstrate off-site power system design and licensing criteria are maintained. Modifications and setpoint changes will be made to ensure that there is no significant effect on grid stability or reliability. There are no modifications associated with the CPPU, which would increase electrical loads beyond those levels previously included, or revise the logic of the distribution systems.

Station loads under emergency operation/distribution conditions (Emergency Diesel Generators) are based on equipment nameplate data, except for (1) the service water pumps where a conservatively high flow Brake Horsepower (BHP) is used, and (2) the CS pumps where the BHP at maximum pump runout is used (Note that the BHP at maximum pump runout is significantly less than

nameplate BHP). CPPU conditions are achieved by utilizing existing equipment as described above. Therefore, under emergency conditions, the electrical supply and distribution components are adequate.

[[

]] The systems have sufficient capacity to support all required loads to achieve and maintain safe shutdown conditions and to operate the ECCS equipment following postulated accidents and transients.

6.1.2 AC Power (normal operation)

The VYNPS on-site power distribution system consists of transformers, numerous buses, and switchgear. AC power to the distribution system is provided from the transmission system or from onsite Diesel Generators. Station batteries provide Direct Current (DC) power to the distribution system.

The on-site power distribution system loads were reviewed under normal, transient, and emergency operating scenarios. In all cases, loads were computed based on equipment nameplate ratings or BHP. These loads were used as inputs for the computation of load, voltage drop, and short circuit current values. Plant operation at the CPPU RTP level for normal, transient, and emergency conditions results in equipment within its allowable temperature rise conditions. Condensate and Reactor Feedwater Pump (RFP) motors will be evaluated for operation in the higher summer temperatures at CPPU. Additional operational loads and increases resulting from CPPU have been evaluated. Bus loading, voltages and fault current levels are within system limits.

Station loads under normal operation/distribution conditions were computed based on equipment nameplate data or BHP with conservative demand factors applied. The only significant change in electrical load demand is associated with power generation system pump motors where three RFPs are required for normal operation at CPPU RTP (at CLTP, two RFPs are operated with one pump/motor in standby). The FW system experiences increased flow demand due to CPPU conditions. A small increase in load occurs with the Condensate Pumps. These motors operate within their allowable temperature rise for the CPPU loading conditions. Load increases do not overload the station buses or result in bus voltages below acceptable limits. The environmental qualification of system equipment will be maintained as discussed in Section 10.3.1. Operation at the CPPU RTP is achieved by utilizing existing distribution equipment operating at or below the nameplate rating; therefore, under normal conditions, the electrical supply and distribution components (switchgear, MCCs, cables, etc.) are adequate.

Station loads under emergency operation/distribution conditions (Emergency Diesel Generators) are based on equipment nameplate data, except for (1) the service water pumps where a conservatively high flow BHP is used, and (2) the CS pumps where the BHP at maximum pump runout is used

(Note that the BHP at maximum pump runout is significantly less than nameplate BHP). CPPU conditions are achieved by utilizing existing equipment as described above. Therefore, under emergency conditions the electrical supply and distribution components are adequate.

[[

]] The systems have sufficient capacity to support all required loads for safe shutdown, to maintain a safe shutdown condition, and to operate the engineered safety feature equipment following postulated accidents.

6.2 DC POWER

The VYNPS DC power distribution system provides control and motive power for various systems/components within the plant. The topic addressed in this evaluation is:

Topic	CPPU Disposition	VYNPS Result
DC power requirements	[[	]]

The DC power distribution drawings were reviewed for changes due to the CPPU. Load changes could include DC MOV load increases including NSSS, the generator rewind design change, and any required program changes such as for SBO or Appendix R. There were no identified load changes that affect the DC power system, therefore the battery duty cycle, voltages to end devices, and available fault currents are within the design as documented in the existing calculations and specifications.

Operation at the CPPU conditions does not increase any load beyond that credited in the existing calculations or revise any component operating duty cycle. Therefore, the DC power distribution system remains adequate.

6.3 FUEL POOL

The VYNPS fuel pool systems consist of storage pools, fuel racks, the Fuel Pool Cooling and Demineralizer System (FPCDS), and the RHR system Augmented FPC mode. The FPCDS consists of two subsystems: a non-safety Normal Fuel Pool Cooling Subsystem (NFPCS) and a safety-related SFPCS. The objective of the FPCDS is to remove the decay heat from the fuel assemblies and maintain the fuel pool water within specified temperature limits.

The effects of CPPU on the VYNPS fuel pool are addressed in the following evaluations:

Topic	CPPU Disposition	VYNPS Result
6.3.1 Fuel Pool Cooling (normal core offload)	[[	
6.3.1 Fuel Pool Cooling (full core offload)		
6.3.2 Crud Activity and Corrosion Products		
6.3.3 Radiation Levels		
6.3.4 Fuel Racks		]]

6.3.1 Fuel Pool Cooling

The VYNPS spent fuel pool (SFP) bulk water temperature must be maintained below the licensing limit of 150°F for normal (batch offload) and abnormal (full core offload) conditions. A normal batch offload of approximately 128 fuel bundles is assumed for outage planning. The SFP temperature should be maintained below the administrative limit of 125°F for normal batch offload. The normal batch offload was analyzed (Configuration 1) with both trains each of NFPCS and SFPCS operating, and assumes a single failure of the RHR Augmented FPC. The limiting heat load condition for the SFP assumes all storage locations filled with spent fuel assemblies from prior discharges except for one full core discharge and one batch discharge. The results of the evaluation (Table 6-3) show that the administrative limit is met for the SFP temperature. The batch offload scenario considers an additional bounding case (Configuration 2) where the cooling systems are restricted in the following manner due to an electrical bus failure: the NFPCS has one pump and two heat exchangers available, and the SFPCS has one pump and one heat exchanger, with the RHR Augmented FPC mode unavailable. The licensing limit of 150°F is also met for this case.

The full core offload scenario was analyzed with both trains each of NFPCS and SFPCS in operation and the RHR Augmented FPC mode available, without assuming a single failure. The limiting heat load condition for the SFP assumes all storage locations filled with spent fuel assemblies from prior discharges except for one full core discharge. The full core offload scenario considers two plant configurations. One configuration (Configuration 3) maintains one train of RHR in the shutdown cooling mode for core cooling, and the other configuration (Configuration 4) takes advantage of natural circulation for core cooling without using the RHR loop. The full core offload with natural circulation considers an additional hypothetical case (Configuration 5) where only the RHR Augmented FPC mode is available with all other cooling systems unavailable. The two failures assumed for this configuration is more restrictive than the no failure recommendation for the FPC evaluation in Standard Review Plan (SRP) 9.1.3. The key results of these analyses are also presented in Table 6-3.

The above evaluations assume an 18-month fuel cycle with GE14 fuel, and use ANSI/ANS 5.1-1979 decay heat standards with two-sigma (2 σ) uncertainty.

The CPPU SFP heat load is higher than the pre-CPPU heat load. The CPPU heat loads at the limiting full core offload condition and the normal batch offload are calculated and then the peak bulk pool temperature is determined to evaluate the FPCDS adequacy. CPPU does not affect the heat removal capability of the FPCDS, or the RHR Augmented FPC mode. CPPU results in slightly higher decay heat loads in the core and the SFP during refueling periods. Each reload affects the decay heat generation in the SFP after a batch discharge of fuel from the reactor. The full core offload into the SFP reaches a maximum heat load immediately after the full core discharge. Based on the evaluations of the heat loads and the SFP temperatures, the SFP bulk temperature remains less than 150°F for all configurations except Configuration 5. The analysis using Configuration 5 was performed for informational purposes and assumes a failure scenario beyond that recommended in SRP 9.1.3. Therefore, a peak SFP bulk temperature greater than the 150°F design criteria is acceptable for this configuration.

The SFP normal makeup flow requirement is less than 3 gpm because the evaporation rates are between 0.6 gpm and 2.6 gpm. This can readily be supplied by other available water sources. The normal water makeup capability is not affected by CPPU and remains adequate for CPPU conditions.

In the unlikely event of a complete loss of SFP cooling capability, the SFP would reach the boiling temperature in six hours in the worst case condition after the limiting full core offload. The worst-case boil-off rate would be 90 gpm as indicated by Configuration 4 (see Table 6-3). The Seismic Category I emergency makeup flow capability is at least 250 gpm.

Existing plant instrumentation and procedures provide adequate indications and direction for monitoring and controlling SFP temperature and water level during normal batch offloads and the unexpected case of the limiting full core offload. Symptom based operating procedures exist to provide mitigation strategies including placing additional cooling trains or systems in service, stopping fuel movement, and initiating make-up if necessary. The symptom based entry conditions and mitigation strategies for these procedures do not require changes for CPPU.

6.3.2 Crud Activity and Corrosion Products

The total crud in the SFP increases slightly, assuming that all residual crud in the Reactor Coolant System is transported to the SFP. However, the increase is negligible, and SFP water quality is maintained by the FPCDS.

6.3.3 Radiation Levels

The normal radiation levels around the SFP may increase slightly primarily during fuel handling operation. Current VYNPS radiation procedures and radiation monitoring program would detect any changes in radiation levels and initiate appropriate actions.

6.3.4 Fuel Racks

The increased decay heat from the CPPU results in a higher heat load in the fuel pool during long-term storage. The fuel racks are designed for higher temperatures (212°F) than the licensing limit of 150°F. There is no effect on the design of the VYNPS fuel racks because the original fuel pool design temperature is not exceeded.

[[

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6.4 WATER SYSTEMS

The VYNPS water systems are designed to provide a reliable supply of cooling water for normal operation and DBA conditions. The VYNPS water systems consist of a safety-related Service Water (SW) system with non-safety related portions, the RHR Service Water (RHRSW) system, the main condenser and circulating water system, the Reactor Building Closed Cooling Water (RBCCW) system, the Turbine Building Closed Cooling Water (TBCCW) system, and the Alternate Cooling System (ACS). The topics addressed in this evaluation are:

Topic	CPPU Disposition	VYNPS Result
Water systems performance (safety-related)	[[	
Water systems performance (normal operation)		
Suppression pool cooling (RHR service operation)		
Alternate cooling system (CLTR topic: Ultimate heat sink)		]]

6.4.1 Service Water Systems

The safety-related SW systems include the safety-related SW system and the RHRSW system. The safety-related SW system individually and in conjunction with the RHRSW system provides cooling water to safety-related Core Standby Cooling system components during accident conditions including events where normal power is lost. The SW system also provides cooling to the SFPCS during normal operation, following a DBA concurrent with a loss of offsite power and a single failure, and when high decay heat loads are placed in the SFP. The ACS provides an alternate means of cooling in the unlikely event that the SW pumps become inoperable. The non-safety related SW systems include portions of the SW system, the RBCCW system (Section 6.4.3), and the TBCCW system (Section 6.4.4).

6.4.1.1 Safety-Related Portions of the SW System

The safety-related SW system provides cooling water from the Connecticut River, which is the ultimate heat sink (UHS). It is designed to provide a reliable supply of cooling water during and following a DBA for the following essential equipment and systems:

- RHR Heat Exchangers
- SFPCS Heat Exchangers
- Emergency Diesel Generator Coolers
- ECCS Room Coolers
- RHRSW Pump Motor Coolers

RBCCW is also supplied by the safety-related portion of the SW system, but it is not an essential system for DBAs. The performance of the safety-related SW system during and following the most demanding design basis event, the LOCA, was demonstrated. Adequate SW system heat transfer capabilities exist at CPPU to support the above components. In addition, the SW flow rates do not change.

6.4.1.2 Non-Safety Related Portions of the SW System

The performance of the non-safety related portions of the SW system at CPPU is adequate to provide the required heat removal and flow requirements to the following affected components:

- TBCCW Heat Exchangers (see Section 6.4.4)
- Generator H_2 Coolers, Stator Water Coolers, and Altrex Coolers
- SFPC Heat Exchangers as needed for supplemental cooling
- Turbine Building Coolers and Condensate Pump Area Coolers

6.4.1.3 RHRSW System

The RHRSW system is normally supplied by the safety-related portion of the SW system. The containment cooling analysis in Section 4.1.1 shows that the post-LOCA RHR heat load increases due to an increase in the maximum suppression pool temperature that occurs following a LOCA. The post-LOCA containment and suppression pool responses have been calculated based on an energy balance between the post-LOCA heat loads and the existing heat removal capacity of the RHR and RHRSW systems. As discussed in Sections 3.5.2 and 4.1.1, the existing suppression pool structure and associated equipment have been reviewed for acceptability based on this increased post-LOCA suppression pool temperature. The containment cooling analysis and equipment review demonstrate that the suppression pool temperature can be maintained within acceptable limits in the post-accident condition at CPPU based on the existing capability of the RHRSW system. Thus, the RHRSW system has sufficient capacity to serve as the coolant supply for long-term core and

containment cooling as required for CPPU conditions. The RHRSW system flow rate is not changed.

6.4.2 Main Condenser/Circulating Water/Normal Heat Sink Performance

The main condenser, circulating water, and heat sink systems are designed to remove the heat rejected to the condenser and thereby maintain adequately low condenser pressure as recommended by the turbine vendor. Maintaining adequately low condenser pressure assures the efficient operation of the turbine-generator and minimizes wear on the turbine last stage buckets.

CPPU operation increases the heat rejected to the condenser and, therefore, reduces the difference between the operating pressure and the recommended maximum condenser pressure. If condenser pressures approach the main turbine backpressure limitation, then reactor thermal power reduction would be required to reduce the heat rejected to the condenser and maintain condenser pressure within the main turbine requirements.

The performance of the main condenser and circulating water system were evaluated for CPPU. This evaluation was based on a design duty over the actual range of circulating water inlet temperatures, and confirms that the condenser, circulating water system, and heat sink are adequate for CPPU operation. Current main turbine backpressure limitations may require load reductions at the upper range of the anticipated circulating water inlet temperatures.

6.4.2.1 Discharge Limits

As shown on Table 6-4, the maximum daily circulating water discharge flow, as specified in the current National Pollutant Discharge Elimination System (NPDES) permit, does not change after CPPU. Changes to the current thermal discharge limits are being proposed to the state authority. These proposed changes will improve condenser performance after CPPU; however, they are not required for CPPU. In any case, the thermal discharge limits defined in the NPDES permit will be maintained after CPPU.

6.4.3 Reactor Building Closed Cooling Water System

The heat loads on the RBCCW system increase < 0.6% as a result of CPPU. The RBCCW heat loads are mainly dependent on the reactor vessel temperature and/or flow rates in the systems cooled by the RBCCW. The change in vessel temperature is minimal and does not result in any significant increase in drywell cooling loads. The flow rates in the systems cooled by the RBCCW (e.g., RRS and RWCU pumps cooling) do not change due to CPPU and, therefore, are not affected by CPPU. The operation of the remaining equipment cooled by the RBCCW (e.g., sample coolers and drain sump coolers) is not power-dependent and is not affected by CPPU. The RBCCW system contains sufficient redundancy in pumps and heat exchangers to assure that adequate heat removal capability is available during normal operation. Sufficient heat removal capacity is available to accommodate the small increase in heat load due to CPPU.

6.4.4 Turbine Building Closed Cooling Water System

The heat load on the TBCCW system, that is power-dependent and that is increased by CPPU, is the heat load from the IPB heat exchanger. A modification to the IPB duct cooling system (consisting of new cooling coils and fans) will be implemented prior to CPPU operation to meet CPPU requirements. In addition, the heat load from the RFP seal and lube oil coolers is increased due to the operation of an additional RFP. These additional heat loads on the TBCCW system are within the current capabilities of the TBCCW system. The CCPU discharge temperatures are below the system design temperature because the TBCCW system is thermally oversized for both the current and CPPU conditions. The additional flows required for the IPB heat exchanger are also within the current capabilities of the TBCCW system.

6.4.5 Alternate Cooling System

During the original licensing of VYNPS, the hypothetical loss of the Vernon Dam was postulated. This led to the design and implementation of the ACS, which has a design inventory of seven days. The ACS is not an engineered safeguards system and is not relied on for DBAs. In addition to the postulated loss of the Vernon Dam, the ACS is credited in the evaluation of two other postulated events: an Appendix R Fire in the intake structure and a flood of the intake structure. The ACS is a heat removal system used in these postulated events to achieve and maintain safe shutdown when the normal SW system (pumping from the Connecticut River) is lost. The ACS was evaluated for CPPU in a manner that is similar to the UHS evaluation for newer plants (e.g., inventory requirements and heat removal capability with increased decay heat).

During ACS operation, the cooling tower and deep basin serve as a heat sink and cooling water supply for required safe shutdown components. The closed-cycle nature of the ACS was evaluated to determine the effect of the higher decay heat on ACS capabilities, specifically the ability to:

- Maintain sufficient water inventory for a seven-day supply of water.

  To meet this design requirement at CPPU, a modification to the RHRSW pump motor bearing cooling water supply line will be implemented to recover the SW that is currently being discharged.

- Assure adequate available NPSH to the RHRSW pumps during the seven-day recirculation mode to the cooling tower basin.

- Provide adequate heat removal to system components during the ACS operating mode.

The heat removal requirements of the following affected components during the ACS operating mode have been evaluated and found to be acceptable at CPPU:

- RHR Heat Exchangers
- SFPCS Heat Exchangers
- Emergency Diesel Generator Coolers
- ECCS Room Coolers

- RHRSW Pump Motor Coolers

6.5 STANDBY LIQUID CONTROL SYSTEM

The SLCS is designed to shut down the reactor from rated power conditions to cold shutdown in the postulated situation that some or all of the control rods cannot be inserted. This manually operated system pumps a highly enriched sodium pentaborate solution into the vessel, to provide neutron absorption and achieve a subcritical reactor condition. SLCS is designed to inject over a wide range of reactor operating pressures. The following topics are addressed in this evaluation:

Topic	CPPU Disposition	VYNPS Result
Core shutdown margin	[[	
System performance and hardware		
Suppression pool temperature following limiting ATWS events		]]

The ability of the SLCS boron solution to achieve and maintain safe shutdown is not a direct function of core thermal power, and therefore, is not affected by the CPPU. SLCS shutdown capability (in terms of the required reactor boron concentration) is reevaluated for each fuel reload. No new fuel product line designs are introduced for CPPU. The boron shutdown concentration of 800 ppm does not change for CPPU. No changes are necessary to the solution volume / concentration or the boron-10 enrichment for CPPU to achieve the required reactor boron concentration for shutdown.

The minimum boron injection rate requirement for maintaining the peak suppression pool water temperature limits, following the limiting ATWS event with SLCS injection, is increased for CPPU.

Using the results of the plant-specific ATWS analysis, the maximum reactor lower plenum pressure following the limiting ATWS event reaches 1292 psia during the time the SLCS is analyzed to be in operation. Consequently, there is a corresponding increase in the maximum pump discharge pressure and a decrease in the operating pressure margin for the pump discharge relief valves. The pressure margin for the pump discharge relief valves remains above the minimum value needed to assure that the relief valves remain closed during system injection. Because of the increase in pump discharge pressure for CPPU, the Technical Specification pump surveillance test pressure (SR 4.4.A.1) will be revised from 1320 psig to 1325 psig.

The SLCS ATWS performance is evaluated in Section 9.3.1 for a representative core design for CPPU. The evaluation shows that CPPU has no adverse effect on the ability of the SLCS to mitigate an ATWS.

6.6 POWER DEPENDENT HVAC

The HVAC systems consist mainly of heating, cooling supply, exhaust, and recirculation units in the turbine building, reactor building, and the drywell. CPPU results in slightly higher process temperatures and small increases in the heat load due to higher electrical currents in some motors and cables. The topic addressed in this evaluation is:

Topic	CPPU Disposition	VYNPS Result
Power dependent HVAC performance	[[	]]

The affected areas are the drywell; the steam tunnel in the reactor building; and the FW heater bay, condenser, and the motor driven condensate and RFP rooms in the turbine building. Other areas in the reactor building and the turbine building are unaffected by the CPPU because the process temperatures remain relatively constant.

The increased heat loads during normal plant operation result in < 1°F increase in the drywell and the main steam tunnel. In the turbine building, the maximum temperature increase in the low pressure and high pressure FW heater areas and the condensate pump room is < 5°F, and the maximum temperature increase in the FW pump room is < 8°F.

The 105°F design ambient room temperature for the condensate and RFP rooms is exceeded in the summer under CPPU conditions. No adverse effect is expected for short periods of time with elevated room temperatures. Affected equipment will be evaluated and dispositioned, as necessary, to assure continued reliable operation at CPPU conditions.

Based on a review of design basis calculations, current area/room temperatures, and CPPU calculations, the design of the HVAC systems are adequate to support CPPU.

6.7 FIRE PROTECTION

This section addresses the effect of CPPU on the fire protection program, fire suppression and detection systems, and reactor and containment system responses to postulated 10 CFR 50, Appendix R fire events. The topics addressed in this evaluation are:

Topic	CPPU Disposition	VYNPS Result
Fire suppression and detection systems	[[	
Operator response time		
Peak cladding temperature		
Vessel water level		
Suppression pool temperature		]]

[[

]] Any changes in physical plant configuration or combustible loading as a result of

modifications to implement the CPPU, will be evaluated in accordance with the plant modification and fire protection programs. The safe shutdown systems and equipment used to achieve and maintain cold shutdown conditions do not change, and are adequate for the CPPU conditions. The operator actions required to mitigate the consequences of a fire are not affected. Therefore, the fire protection systems and analyses are not affected by CPPU.

The reactor and containment response to the postulated 10 CFR 50 Appendix R fire event at CPPU conditions is evaluated in Section 6.7.1. The results show that the peak fuel cladding temperature and containment pressures and temperatures are below the acceptance limits and demonstrate that there is sufficient time available for the operators to perform the necessary actions to achieve and maintain cold shutdown conditions. Therefore, the fire protection systems and analyses are not adversely affected by CPPU.

6.7.1 10 CFR 50 Appendix R Fire Event

A [[ ]] evaluation was performed to demonstrate safe shutdown capability in compliance with the requirements of 10 CFR 50 Appendix R assuming CPPU conditions.

The results of the Appendix R evaluation for CPPU provided in Table 6-5 demonstrate that the fuel cladding integrity and containment integrity are maintained and that sufficient time is available for the operator to perform the necessary actions. The current exemption for the momentary core uncovery during depressurization remains necessary for CPPU. CPPU does not affect any other exemptions described in the VYNPS safe shutdown capability analysis. No changes are necessary to the equipment required for safe shutdown for the Appendix R event. One train of systems remains available to achieve and maintain safe shutdown conditions from either the Main Control Room or the remote shutdown panel. Therefore, CPPU has no adverse effect on the ability of the systems and personnel to mitigate the effects of an Appendix R fire event, and satisfies the requirements of Appendix R with respect to achieving and maintaining safe shutdown in the event of a fire.

6.8 OTHER SYSTEMS AFFECTED BY POWER UPRATE

This section addresses the effect of CPPU on systems not addressed in other sections of this report. The topic addressed in this evaluation is:

Topic	CPPU Disposition	VYNPS Result
Other systems	[[	]]

Based on experience and previous NRC reviews, all systems that are significantly affected by CPPU are addressed in this report. Other systems not addressed by this report are not significantly affected by CPPU.

[[ ]]

Table 6-1
VYNPS CPPU Electrical Characteristics

Parameter	CPPU
Guaranteed Generator Output (MWe)	656
Rated Voltage (kV)	22
Power Factor	0.959
Guaranteed Generator Output (MVA)	684
Generator Current at Rated Voltage (kA)	17.95
Isolated Phase Bus Duct Rating (kA)	19.00
Main Transformers Rating (MVA)	675
CPPU Transformer Output (MVA)	675

Table 6-2
VYNPS Offsite Electric Power System

Component	Rating	CPPU Output
Generator (MVA)	684	684
Isolated Phase Bus Duct (kA)	19.00	17.95 [1.]
Main Transformers (MVA)	675	675
Auxiliary Transformer [2.] (MVA)	39.2	34.4
Start-Up Transformers [3.] (MVA) T-3A T-3B	 28 28	 17.9 24.8
Switchyard (limiting) [4.] (MVA)	675	675

Notes:

1. The IPB Duct current is shown at generator at nominal voltage.
2. The Auxiliary Transformer Rating is the Forced Oil and Air (FOA) Rating at 65°C Rise and the CPPU Output is based on the worst case loading from the Station Load Flow calculation.
3. VYNPS has two Start-Up Transformers. Their Rating is the FOA Rating at 65°C Rise. The CPPU Output is based on the worst case loading from the Station Load Flow calculation.
4. The Switchyard MVA Rating and the CPPU Output is based on the maximum station output through the Generator Step-Up Transformer.

Table 6-3
VYNPS Spent Fuel Pool Parameters for CPPU

Conditions / Parameter	Results	Limit
Configuration 1 – Normal Batch Offload: Both trains each of NFPCS and SFPCS in service, and the heat load in the RPV cooled by the other RHR in SDC mode		
Time to initiate fuel transfer to SFP (hr)	24	≥ 24
Peak SFP Temperature (°F)	116.0	≤ 125
Time to Peak SFP Temperature (hr)	51	N/A
Time to 209.4°F [1.] (hr)	51	N/A
Boil off rate [1.] (gpm)	24.5	N/A
Configuration 2 – Limiting Batch Offload: 1.5 trains of NFPCS and one train of SFPCS in service, and the heat load in the RPV cooled by the other RHR in SDC mode		
Time to initiate fuel transfer to SFP (hr)	24	≥ 24
Peak SFP Temperature (°F)	132.7	≤ 150
Time to Peak SFP Temperature (hr)	62	N/A
Time to 209.2°F [1.] (hr)	47	N/A
Boil off rate [1.] (gpm)	23.7	N/A
Configuration 3 – Full Core Offload: No Failure in the Cooling System, and the heat load in the RPV cooled by the other RHR in SDC mode		
Time to initiate fuel transfer to SFP (hr)	24	≥ 24
Peak SFP Temperature (°F)	122.4	≤ 150
Time to Peak SFP Temperature (hr)	61	N/A
Time to boil (hr)	10	N/A
Boil off rate (gpm)	78	N/A
Configuration 4 – Full Core Offload: No Failure in the Cooling System, and the heat load in the RPV cooled by Natural Circulation		
Time to initiate fuel transfer to SFP (hr)	24	≥ 24
Peak SFP Temperature (°F)	128.6	≤ 150
Time to Peak SFP Temperature (hr)	38	N/A
Time to boil (hr)	9	N/A
Boil off rate (gpm)	90	N/A
Configuration 5 – Full Core Offload: With RHR Augmented FPC mode alone, and the heat load in the RPV cooled by Natural Circulation		
Time to initiate fuel transfer to SFP (hr)	24	≥ 24
Peak SFP Temperature (°F)	159.6 [2.]	N/A
Time to Peak SFP Temperature (hr)	45	N/A
Time for SFP Temperature ≤ 150°F (hr)	3.5	N/A
Time to boil (hr)	6	N/A
Boil off rate (gpm)	87	N/A

Notes:

1. Configurations 1 and 2 never reach 212°F. The boiloff rates are the maximum evaporation rate at the peak SFP temperatures of 209.4°F and 209.2°F, respectively.
2. The peak SFP bulk temperature exceeds 150°F for Configuration 5; this is acceptable because Configuration 5 assumes a failure scenario beyond the SRP 9.1.3 recommendation.

Table 6-4
VYNPS Effluent Discharge Comparison

Parameter	State Limit	Maximum Current [1]	CPPU
Flow [2] (million gallons/day)	543	532 [3]	Unchanged
Downstream (Station 3) Temperature 24-hour avg. (°F)	Summer – none Winter - 65	Summer – 81.02 Winter – 52.03	Unchanged
Downstream (Station 3) Temperature 1-hour avg. (°F)	Summer – none Winter – 65	Summer – 82.37 Winter – 57.40	Unchanged
In-stream ΔT 24-hour avg. [4] (°F) (Depends on upstream ambient conditions, and is largely affected by river flow)	Summer –2 3 4 5 Winter – 13.4 [5]	Summer – 1.93 2.02 1.78 1.26 Winter – 12.71	Unchanged [6]
Chlorine (average/day) / (mg/L/per day)	2 hrs/day / 0.2 mg/L/day	2 hrs/day / 0.05 mg/L/day	Unchanged

Notes:

1. Values are based on 2002 data only, most conservative value for each observed parameter.
2. Based on current pump design capacity, which is not changing.
3. The 532 mgal/day is taken from April 2002 during open cycle (highest daily discharge). The maximum state limit for flow during closed cycle is 12.1 mgal/day.
4. During the period May 16 to October 14, the increase in temperature above upstream ambient shall not exceed the allowable ΔT limits, which are based on upstream ambient conditions.
5. Under existing NPDES Permit conditions, this will require cooling tower use above 106% power and with minimum river flow (1,250 cfs). At 120% power with the existing 13.4°F ΔT winter limit, VY would have to down power or use cooling towers if river flow dropped below 1,500 cfs.
6. In addition to the 13.4°F ΔT winter limit, the NPDES Permit also indicates that the rate of change in temperature at the downstream Station 3 cannot exceed 5°F per hour and the temperature at Station 3 shall never exceed 65°F.

Table 6-4 (Cont.)
Proposed Summer NPDES Permit Change

Upstream River Temperature (Station 7)	Existing ΔT Limit	Proposed ΔT Limit
Above 78°F	2	2
Greater than 63°F, Less than or equal to 78°F	2	3
Greater than 59°F, Less than or equal to 63°F	3	4
Greater than or equal to 55°F, Less than or equal to 59°F	4	5
Below 55°F	5	5

Note:

1. The current NPDES allowable temperature increase is a function of the upstream river temperature, as shown. Although not required for CPPU, a proposed change to this summer NPDES permit condition has been submitted to the state authority. The winter NPDES permit condition remains as is.

Table 6-5
VYNPS Appendix R Fire Event Evaluation Results

	CLTP	CPPU	Appendix R Criteria
Time to Core Uncovery (minutes)	25.3	21.3	≤ 21.3 [1.]
Cladding Heatup (PCT) (°F)	1292.9	1475.4	≤ 1500
Peak Drywell Pressure (psig)	23.6	23.6	≤ 25 [2.]
Suppression Pool Bulk Temperature (°F)	180.9	189.5	≤ 281 [3.] ≤ 195 [4.]
Net Positive Suction Head [5.]	Yes	Yes [6.]	Adequate for system using suppression pool water source

Notes:

1. Time required to initiate RCIC.
2. Drywell design pressure is 56 psig. The Appendix R Criterion is based on the RCIC operational limit.
3. Containment structure design limit.
4. Torus attached piping limit.
5. NPSH demonstrated adequate, see Section 4.2.
6. Overpressure credit required, see Section 4.2.6.

7. POWER CONVERSION SYSTEMS

This section primarily focuses on the information requested in RG 1.70, Chapter 10, that applies to CPPU.

7.1 TURBINE-GENERATOR

The VYNPS turbine-generator converts the thermal energy in the steam into electrical energy. The topics addressed in this evaluation are:

Topic	CPPU Disposition	VYNPS Result
Turbine-generator performance	[[	
Turbine-generator missile avoidance		]]

The turbine and generator was originally designed with a maximum flow-passing capability and generator output in excess of rated conditions to ensure that the original rated steam-passing capability and generator output is achieved. This excess design capacity ensures that the turbine and generator meet rated conditions for continuous operating capability with allowances for variations in flow coefficients from expected values, manufacturing tolerances, and other variables that may adversely affect the flow-passing capability of the unit. The difference in the steam-passing capability between the design condition and the rated condition is called the flow margin.

The turbine-generator was originally designed and operates with a flow margin of 5%. The current rated throttle steam flow is 6.456 Mlb/hr at a throttle pressure of 983 psia. The generator at CLTP is rated at 626 MVA, which results in a rated electrical output (gross) of 563 MWe at a power factor of 0.9.

At the CPPU RTP and reactor dome pressure of 1025 psia, the turbine operates at an increased rated throttle steam flow of 7.900 Mlb/hr and at a throttle pressure of 961 psia. To maintain control capability, GE uses a minimum target value of approximately 5% throttle flow margin, with controllability confirmed by unit testing as described in Section 10.4. For operation at CPPU, the high-pressure turbine has been redesigned with a new rotor, diaphragms, and buckets for at least the minimum target throttle flow margin, to increase its flow passing capability. The entire turbine steam path and turbine auxiliary systems have been evaluated at the CPPU RTP and have been found acceptable.

In order to achieve a higher target gross electrical output at CPPU RTP while continuing to allow the VYNPS to dispatch reactive power, a generator stator rewind will be performed prior to CPPU RTP implementation. The stator rewind allows a re-rate of the VYNPS generator to a rated electrical output of 656 MWe at a power factor of 0.96. Revised generator reactive capability curves at CPPU conditions maintain the generator stator core and field winding within their design limits, i.e., no modification to the stator core and field winding is required for

CPPU. The generator hydrogen cooling system pressure is unchanged at CPPU RTP. However, the hydrogen cooling system heat exchangers are replaced by heat exchangers of higher capacity due to the increased heat removal requirements at CPPU RTP. The generator and generator auxiliaries have been evaluated at CPPU RTP and have been found acceptable.

The high-pressure and low-pressure turbine rotors at VYNPS (for both CLTP and CPPU RTP) have integral, non-shrunk on wheels. Per CLTR Section 7.1, a separate rotor missile analysis is not required for plants with integral wheels. The overspeed calculation compares the entrapped steam energy contained within the turbine and the associated piping, after the stop valves trip, and the sensitivity of the rotor train for the capability of overspeeding. The entrapped energy increases slightly for the CPPU conditions. The hardware modification design and implementation process establishes the overspeed trip settings to provide protection for a TT.

7.2 CONDENSER AND STEAM JET AIR EJECTORS

The VYNPS condenser converts the steam discharged from the turbine to water to provide a source for the condensate and feedwater systems. The Steam Jet Air Ejectors (SJAEs) remove noncondensable gases from the condenser to improve thermal performance. The topic addressed in this evaluation is:

Topic	CPPU Disposition	VYNPS Result
Condenser and SJAE	[[	]]

The condenser and SJAE functions are required for normal plant operation and are not safety-related.

The condenser thermal performance evaluation at CPPU conditions determined that the 5" Hg_a turbine backpressure limit can be satisfied with a cold water inlet temperature of 90°F. This evaluation assumed an apparent cleanliness as low as 67%, which includes consideration for current condenser tube plugging of approximately 3.7%.

Condenser hotwell capacities and level instrumentation are adequate for CPPU conditions. The potential for excessive condenser tube vibration, at the higher CPPU steam flow, is being addressed by the installation of anti-vibration tube stakes. Existing maintenance programs include eddy current testing, condenser tubes measurements, and general material condition inspections.

The design of the condenser air removal system is not adversely affected by CPPU and no modification to the system is required. The physical size of the primary condenser and evacuation time are the main factors in establishing the capabilities of the vacuum pumps. These parameters do not change. Because flow rates do not change, there is no change to the holdup time in the pump discharge line routed to the station stack. The capacity of the SJAEs is adequate because they were originally designed for operation at flows greater than those required at CPPU conditions.

7.3 TURBINE STEAM BYPASS

The Turbine Steam Bypass System provides a means of accommodating excess steam generated during normal plant maneuvers and transients. The topics addressed in this evaluation are:

Topic	CPPU Disposition	VYNPS Result
Turbine steam bypass (normal operation)	[[	
Turbine steam bypass (safety analysis)		]]

The turbine bypass valves were initially rated for a total steam flow capacity of not less than 105% of the original rated reactor steam flow, or ~7.06 Mlb/hr. Each of ten bypass valves is designed to pass a steam flow of ~706,000 lbm/hr and does not change at CPPU RTP. At CPPU conditions, rated reactor steam flow is 7.906 Mlb/hr, resulting in a bypass capacity of 89% of CPPU rated steam flow. The bypass capacity at VYNPS remains adequate for normal operational flexibility at CPPU RTP.

The bypass capacity is used as an input to the reload analysis process for the evaluation of limiting events that credit the Turbine Steam Bypass System (see Section 9.1). [[

]]

7.4 FEEDWATER AND CONDENSATE SYSTEMS

The Feedwater and Condensate Systems provide the source of makeup water to the reactor to support normal plant operation. The topic addressed in this evaluation is:

Topic	CPPU Disposition	VYNPS Result
Feedwater and condensate systems	[[	]]

The FW and Condensate systems do not perform a system level safety-related function, and are designed to provide a reliable supply of FW at the temperature, pressure, quality, and flow rate as required by the reactor. However, their performance has a major effect on plant availability and capability to operate at the CPPU conditions. For CPPU, the FW and Condensate systems will meet their performance requirements with the following operational change and modification:

- Operational change: The existing spare RFP will be placed in operation to support the normal CPPU system flow and head requirements.
- Plant modification: The high pressure heaters will be replaced to ensure reliable operation and support the higher extraction pressures associated with CPPU.

7.4.1 Normal Operation

System operating flows at CPPU increase to approximately 123% of rated flow at the CLTP. The FW and Condensate systems at CPPU require operation of the current standby RFP to assure acceptable performance with the new system operating conditions.

The low pressure FW heaters located upstream of the RFPs were analyzed and verified to be acceptable for the higher CPPU FW heater flows, temperatures, and pressures. The currently installed high pressure FW heaters located at the discharge of the RFPs do not meet CPPU design pressure requirements and will therefore be replaced. The thermal performance of the FW heaters will be monitored during the CPPU power ascension program.

7.4.2 Transient Operation

With the CPPU three RFP operating configuration, the FW and Condensate systems have the capability to supply approximately 134% of the CLTP rated flow requirements. This provides a 9% margin above required CPPU rated flow. For system operation with all system pumps available, the predicted operating parameters are acceptable and within the component capabilities.

The post RFP trip system capacity was evaluated to confirm that for the FW and Condensate system configurations, the capability to supply the transient flow requirements is maintained.

Additionally, as a trip avoidance enhancement, a reactor recirculation system runback modification will be installed to avoid a plant trip on a loss of a condensate pump or RFP. To provide additional operating margin and avoid a trip of the RFP on the loss of a condensate pump, the RFP suction pressure trip setpoint will be lowered and will continue to have acceptable margin for the RFP NPSH required.

7.4.3 Condensate Demineralizers

The effect of CPPU on the Condensate Filter Demineralizers (CFDs) was reviewed. The CFD system requires modification to support CFD full flow operation during backwash and pre-coating without requiring a plant power reduction. This modification consists of adding a filtered bypass around the condensate demineralizer system to allow for the removal of one condensate demineralizer element during the periodic backwashing and pre-coating process as well as for general maintenance or element replacement. As a result of the CPPU, the system experiences slightly higher loadings resulting in slightly reduced CFD run times. However, the reduced run times are acceptable (refer to Section 8 for the effect on the radwaste systems).

8. RADWASTE AND RADIATION SOURCES

This section primarily focuses on the information requested in RG 1.70, Chapters 11 and 12, that applies to CPPU.

8.1 LIQUID AND SOLID WASTE MANAGEMENT

The VYNPS Liquid and Solid Radwaste System collects, monitors, processes, stores, and returns processed radioactive waste to the plant for reuse or for discharge. The topics addressed in this evaluation are:

Topic	CPPU Disposition	VYNPS Result
Coolant fission and corrosion product levels	[[	
Waste Volumes		]]

The single largest source of liquid and wet solid waste contributing to an increase due to CPPU is from the backwash of condensate demineralizers. CPPU results in an increased flow rate through the condensate demineralizers, resulting in a reduction in the average time between backwashes. This reduction does not affect plant safety. Similarly, the RWCU filter-demineralizers require more frequent backwashes due to higher levels of impurities as a result of the increased FW flow.

The floor drain collector subsystem and the waste collector subsystem both receive periodic inputs from a variety of sources. CPPU does not affect system operation or equipment performance. Therefore, neither subsystem is expected to experience a large increase in the total volume of liquid and solid waste due to operation at the CPPU condition.

The increased loading of soluble and insoluble species increases the volume of the liquid processed wastes by 1.2% and that of the solid processed wastes by 17.8%. The total volume of liquid and solid processed waste does not increase appreciably (as compared to the Radwaste System capacity) because the only increase in processed waste is due to more frequent backwashes of the condensate demineralizers and RWCU filter demineralizers. The total liquid and solid increases are within the Radwaste System capacity. Therefore, CPPU does not have an adverse effect on the processing of liquid and solid radwaste.

8.2 GASEOUS WASTE MANAGEMENT

The topics addressed in this evaluation are:

Topic	CPPU Disposition	VYNPS Result
Offsite release rate	[[	
Recombiner performance		]]

The primary function of the Gaseous Waste Management (Offgas) System is to process and control the release of gaseous radioactive effluents to the site environs so that the total radiation exposure of persons in offsite areas is within the guideline values of 10 CFR 50, Appendix I.

The radiological release rate is administratively controlled to remain within existing site release rate limits, and is a function of fuel cladding performance, main condenser air inleakage, charcoal adsorber inlet dew point, and charcoal adsorber temperature. [[

]] Thus, the recombiner and condenser, as well as downstream system components, are designed to handle an average increase in thermal power of as much as 70% relative to CLTP, without exceeding the design basis temperatures, flow rates, or heat loads. Therefore, the gaseous radwaste system at VYNPS is confirmed to be consistent with GE design specifications for radiolytic flow rate [[
]]

8.3 RADIATION SOURCES IN THE REACTOR CORE

During power operation, the radiation sources in the core are directly related to the fission rate. These sources include radiation from the fission process, accumulated fission products and neutron reactions as a secondary result of fission. Historically, these sources have been defined in terms of energy or activity released per unit of reactor power. Therefore, for a CPPU, the percent increase in the operating source terms is no greater than the percent increase in power. The topic addressed in this evaluation is:

Topic	CPPU Disposition	VYNPS Result
Post operational radiation sources for radiological and shielding analysis	[[	]]

The post-operation radiation sources in the core are primarily the result of accumulated fission products. Two separate forms of post-operation source data are normally applied. The first of these is the core gamma-ray source, which is used in shielding calculations for the core and for individual fuel bundles. This source term is defined in terms of MeV/sec per Watt of reactor thermal power (or equivalent) at various times after shutdown. The total gamma energy source, therefore, increases in proportion to reactor power.

The second set of post-operation source data consists primarily of nuclide activity inventories for fission products in the fuel. These data are needed for post-accident and SFP evaluations, which are performed in compliance with regulatory guidance that applies different release and transport assumptions to different fission products. The core fission product inventories for these evaluations are based on an assumed fuel irradiation time, which develops "equilibrium" activities in the fuel (typically 3 years). Most radiologically significant fission products reach equilibrium within a 60-day period. [[

]] The radionuclide inventories are provided in terms of Curies per megawatt of reactor thermal power at various times after shutdown.

The VYNPS specific parameters are enveloped by the bounding parameters of the radiation sources in the reactor core generic description provided in the CLTR. The results of the VYNPS plant-specific radiation sources evaluation are included in the LOCA, FHA, and CRDA radiological analyses presented in Section 9.2. A plant-specific analysis for NUREG-0737, Item II.B.2, post-accident mission doses was performed in which the evaluated mission doses for VYNPS are demonstrated to be less than 5 rem TEDE. Details of the analysis are contained in the AST submittal (Reference 27), which describes the full implementation of the AST methodology at CPPU conditions.

8.4 RADIATION SOURCES IN REACTOR COOLANT

Radiation sources in the reactor coolant at VYNPS include activation products and activated corrosion and fission products. The topics addressed in this evaluation are:

Topic	CPPU Disposition	VYNPS Result
8.4.1 Coolant Activation Products	[[	
8.4.2 Activated Corrosion and Fission Products		]]

8.4.1 Coolant Activation Products

During reactor operation, the coolant passing through the core region becomes radioactive as a result of nuclear reactions. The coolant activation, especially N-16 activity, is the dominant source in the turbine building and in the lower regions of the drywell. The activation of the water in the core region is in approximate proportion to the increase in thermal power. [[

]]

8.4.2 Activated Corrosion Products and Fission Products

The reactor coolant contains activated corrosion products, which are the result of metallic materials entering the water and being activated in the reactor region. Under the CPPU conditions, the feedwater flow increases with power and the activation rate in the reactor region increases with power. The net result is an increase in the activated corrosion product production.

Fission products in the reactor coolant are separable into the products in the steam and the products in the reactor water. The activity in the steam consists of noble gases released from the core plus carryover activity from the reactor water. This activity is the noble gas offgas that is included in the plant design. The calculated offgas rates for CPPU after thirty minutes decay are well below the original design basis of 0.03 curies/sec. Therefore, no change is required in the design basis for offgas activity for the CPPU.

The fission product activity in the reactor water, like the activity in the steam, is the result of minute releases from the fuel rods. Fission product activity levels in the reactor water at design carry over rates were calculated to be less than the design basis, therefore requiring no change.

8.5 RADIATION LEVELS

For CPPU at VYNPS, normal operation radiation levels increase by approximately the percentage increase in power level. Some areas reflect an additional small increase due to accelerated steam flow. For conservatism, many aspects of the plant were originally designed for higher-than-expected radiation sources. Thus, the increase in radiation levels does not affect radiation zoning or shielding in the various areas of the plant because it is offset by conservatism in the original design, source terms used, and analytical techniques. The topics addressed in this evaluation are:

Topic	CPPU Disposition	VYNPS Result
Normal operational radiation levels	[[	
Post-operation radiation levels		
Post-accident radiation levels		]]

The normal operating radiation levels specified for VYNPS are generally based on dose rate measurements at various locations during plant operation at CLTP conditions. The normal operating radiation levels specified for CLTP conditions were evaluated to increase in proportion to the increase in thermal power. The increased normal radiation levels were evaluated and determined to have no adverse effect on safety-related plant equipment as indicated in Sections 10.3.1 and 10.3.2. Individual worker exposures can be maintained within acceptable limits by controlling access to radiation areas in conjunction with procedural controls and the site ALARA (As Low as Reasonably Achievable) program. In addition, VYNPS has previously implemented noble metal chemical addition to limit the increase in normal radiation doses from the implementation of hydrogen water chemistry.

[[

]] Regardless, individual worker exposures can be maintained within acceptable limits by controlling access to radiation areas using the site ALARA program. Procedural controls compensate for increased radiation levels. Radiation measurements will be made at selected power levels to ensure the protection of personnel.

Post-accident radiation levels were evaluated for radiological consequences using the RG 1.183 AST methodology, as part of the VYNPS plant-specific accident analyses presented in Section 9.2. Accident radiation levels at CLTP were evaluated using the TID source term methodology. Post-accident radiation levels remain below established regulatory limits for CPPU conditions. Details of the accident radiological analysis are contained in a separate VYNPS LAR (Reference 27) describing full implementation of the AST methodology at CPPU conditions. The increased post-accident radiation doses have no adverse effect on safety-related plant equipment as indicated in Sections 10.3.1 and 10.3.2. A plant-specific analysis for NUREG-0737, Item II.B.2, post-accident mission doses has been performed, the details of which are provided in the AST LAR (Reference 27).

Section 9.2 addresses the accident doses for the Control Room.

8.6 NORMAL OPERATION OFF-SITE DOSES

The primary sources of normal operation offsite doses at VYNPS are airborne releases from the Offgas System and gamma shine from the plant turbines. The topics addressed are:

Topic	CPPU Disposition	VYNPS Result
Plant gaseous emissions	[[	
Plant skyshine from the turbine		]]

The increase in normal operation activity levels in the reactor coolant is proportional to the percentage increase in core thermal power, i.e., 20%. Noble gas levels in the steam phase are expected to be approximately the same as pre-CPPU conditions because the increase in steaming rate is approximately the same as the production rate due to CPPU. Noble gas release through the off-gas system and release of tritium is conservatively estimated to increase proportionally to the CPPU. Steam activity levels for species related to carryover (halogens & particulates) and volatile halogens will increase proportionally to changes in reactor coolant and the moisture carryover fraction. Examination of the normal operation radiological effluent doses reported for the last five years (1997-2001) indicates that the estimated doses due to the pre-CPPU gaseous releases (~1 mrem) are a very small fraction of the 10CFR 50 Appendix I guidelines; and that there were no radiological liquid effluents discharged during this time period. While the normal operation releases and doses are expected to increase due to CPPU, the dose effect remains well within the limits of 10 CFR 20, 10 CFR 50, Appendix I, and 40 CFR 190.

VYNPS has previously implemented noble metal chemical addition to limit the increase in normal radiation doses from the future implementation of hydrogen water chemistry. The N-16 activity level in reactor steam during CPPU increases by approximately the percentage of the uprate due to a higher partition factor to the steam phase that is caused by the increase of the core flow boiling fraction. The CPPU increase in steam flow results in higher levels of N-16 and other activation products in the turbines. The increased flow rate and velocity, which result in shorter travel times to the turbine and less radioactive decay in transit, lead to higher radiation levels in and around the turbines and offsite skyshine dose. The overall CPPU increase factor for the N-16 source in the Turbine Building is estimated to be 26%. Based on currently reported pre-CPPU doses, the maximum site boundary annual dose due to direct and skyshine from all plant sources for CPPU, and assuming implementation of hydrogen water chemistry, is estimated to be 18.6 mrem and therefore, remain within the state and federal regulatory limits.

9. REACTOR SAFETY PERFORMANCE EVALUATIONS

This section primarily focuses on the information requested in RG 1.70, Chapter 15, which applies to CPPU.

9.1 ANTICIPATED OPERATIONAL OCCURRENCES

The AOO events include fuel thermal margin and loss of water level events. The overpressure protection analysis events are addressed in Section 3.1 of this report. The topics addressed in this evaluation are:

Topic	CPPU Disposition	VYNPS Result
9.1.1 Fuel Thermal Margins Events	[[	
9.1.2 Power and Flow Dependent Limits		
9.1.3 Loss of Water Level Events (Loss of feedwater flow)		
9.1.3 Loss of Water Level Events (Loss of one feedwater pump)		]]

9.1.1 Fuel Thermal Margin Events

[[

]]

9.1.2 Power and Flow Dependent Limits

The operating limit MCPR, LHGR, and/or MAPLHGR thermal limits are modified by a flow factor when the plant is operating at less than 100% core flow. This flow factor is primarily based upon an evaluation of the slow recirculation increase event. [[

]]

Similarly, the thermal limits are modified by a power factor when the plant is operating at less than 100% power. [[

]]

9.1.3 Loss of Water Level Events

For the LOFW event, adequate transient core cooling is provided by maintaining the water level inside the core shroud above the top of active fuel. A plant-specific analysis was performed for VYNPS at CPPU conditions. This analysis assumed failure of the HPCI system and used only the RCIC system to restore the reactor water level. Because of the extra decay heat from CPPU, slightly more time is required for the automatic systems to restore water level. Operator action is only needed for long-term plant shutdown. The results of the LOFW analysis for VYNPS show that the minimum water level inside the shroud is 80 inches above the top of active fuel at CPPU conditions. After the water level is restored, the operator manually controls the water level, reduces reactor pressure, and initiates RHR shutdown cooling. This sequence of events does not require any new operator actions or shorter operator response times. Therefore, the operator actions for a LOFW transient do not significantly change for CPPU.

As discussed in Section 3.9, an operational requirement is that the RCIC system restores the reactor water level while avoiding the level at which the operators would manually initiate the ADS system. This requirement is intended to avoid unnecessary initiations of safety systems. This requirement is not a safety-related function. The results of the LOFW analysis for VYNPS show that this operational requirement is met.

The loss of one RFP event only addresses operational considerations to avoid reactor scram on low reactor water level. This requirement is intended to avoid unnecessary reactor shutdowns. Because the MELLLA region is extended along the existing upper boundary to the CPPU RTP,

there is no increase in highest flow control line for the VYNPS CPPU. [[

]]

9.2 DESIGN BASIS ACCIDENTS

This section addresses the radiological consequences of DBAs for VYNPS. The topics addressed in this evaluation are:

Topic	CPPU Disposition	VYNPS Result
Main Steam Line Break Outside Containment	[[	
Instrument Line Break		
LOCA Inside Containment		
Fuel Handling Accident		
Control Rod Drop Accident		]]

The magnitude of radiological consequences of a DBA is proportional to the quantity of radioactivity released to the environment. This quantity is a function of the fission products released from the core as well as the transport mechanism between the core and the release point.

VYNPS has submitted an LAR (Reference 27) describing full implementation of the AST methodology, at CPPU conditions, that complies with Regulatory Guide 1.183. This methodology has been used in the evaluation of DBA radiological consequences.

The Main Steam Line Break Accident (MSLBA) analysis for VYNPS is based on hot standby conditions and [[
]] Therefore, the resulting radiological consequences remain within applicable regulatory criteria for the MSLBA at CPPU conditions.

The Instrument Line Break (ILB) is not considered a DBA for VYNPS.

For the LOCA inside Containment, FHA, and CRDA, the whole body and thyroid doses were calculated at the exclusion area boundary, Low Population Zone (LPZ), and in the Control Room. The doses resulting from the accidents analyzed are compared with the applicable dose limits in Tables 9-1 through 9-3, for both the CPPU and pre-CPPU RTP levels. The effect of extended burnup on the FHA was not evaluated, per RG 1.183, based on CPPU core average bundle power of 5.3 MWt and peak exposure of 58 GWD/MT. The [[ ]] results for the CPPU remain below established regulatory limits.

9.3 SPECIAL EVENTS

This section considers two special events: ATWS and SBO. The topics addressed in this evaluation are:

Topic	CPPU Disposition	VYNPS Result
9.3.1 ATWS (Overpressure) - Event Selection	[[	
9.3.1 ATWS (Overpressure) - Limiting Events		
9.3.1 ATWS (Suppression Pool Temperature) - Event Selection		
9.3.1 ATWS (Suppression Pool Temperature) - Limiting Events		
9.3.1 ATWS (Peak Cladding Temperature)		
9.3.2 Station Blackout		
9.3.3 ATWS with Core Instability		]]

9.3.1 Anticipated Transient Without Scram

The overpressure evaluation includes consideration of the most limiting RPV overpressure case. [[

]]

For VYNPS, the LOOP does not result in a reduction in the RHR pool cooling capability relative to the MSIVC and PRFO cases. With the same RHR pool cooling capability, the containment response for the MSIVC and PRFO cases bound the LOOP case. [[

]]

VYNPS meets the ATWS mitigation requirements defined in 10 CFR 50.62:

1. Installation of an Alternate Rod Insertion (ARI) system;
2. Boron injection equivalent to 86 gpm; and
3. Installation of automatic Recirculation Pump Trip (RPT) logic (i.e., ATWS-RPT).

In addition, plant-specific ATWS analyses were performed to ensure that the following ATWS acceptance criteria are met:

1. Peak vessel bottom pressure less than ASME Service Level C limit of 1500 psig;

2. Peak suppression pool temperature less than 281°F (Wetwell shell design temperature); and
3. Peak containment pressure less than 62 psig (110% of drywell design pressure).

The limiting events for the acceptance criteria discussed above are the PRFO event and the MSIVC event.

The ATWS analyses were performed for CLTP and for CPPU RTP to demonstrate the effect of the CPPU on the ATWS acceptance criteria. There is no change to the required hot shutdown boron weight for the CPPU ATWS analysis. The key inputs to the ATWS analysis are provided in Table 9-4. The results of the analysis are provided in Table 9-5.

The results of the ATWS analyses meet the above ATWS acceptance criteria. Therefore, the VYNPS response to an ATWS event at CPPU is acceptable.

Coolable core geometry is assured by meeting the 2200°F peak cladding temperature and the 17% local cladding oxidation acceptance criteria of 10 CFR 50.46. [[

]]

9.3.2 Station Blackout

SBO was reevaluated using the guidelines of NUMARC 87-00. The plant response to and coping capabilities for an SBO event are affected slightly by operation at CPPU RTP, due to the increase in the initial power level and decay heat. Decay heat was conservatively evaluated assuming end-of-cycle (18-month) and GE14 fuel. There are no changes to the systems and equipment used to respond to an SBO, nor is the required coping time changed.

Areas containing equipment necessary to cope with an SBO event were evaluated for the effect of loss-of-ventilation due to an SBO. The evaluation shows that equipment operability is bounded due to conservatism in the existing design and qualification bases. The battery capacity remains adequate to support HPCI/RCIC operation after CPPU. Adequate compressed gas capacity exists to support the SRV actuations.

The current CST inventory reserve and restoration of Alternate AC within 10 minutes ensures that adequate water volume is available to remove decay heat, depressurize the reactor, and maintain reactor vessel level above the top of active fuel. Consistent with the DBA-LOCA condition, the required NPSH margin for the RHR pumps has been evaluated (see Section 4.2.6) and a component acceptability review has been completed (see Section 3.10).

Based on the above evaluations, VYNPS continues to meet the requirements of 10 CFR 50.63 after the CPPU.

9.3.3 ATWS with Core Instability

The ATWS with core instability event occurs at natural circulation following a recirculation pump trip. Therefore, it is initiated at approximately the same power level as a result of CPPU operation because the MELLLA upper boundary is not increased. The core design necessary to achieve CPPU operations may affect the susceptibility to coupled thermal-hydraulic/neutronic core oscillations at the natural circulation condition, but would not significantly affect the event progression.

Several factors affect the response of an ATWS instability event, including operating power and flow conditions and core design. The limiting ATWS core instability evaluation presented in References 28 and 29 was performed for an assumed plant initially operating at CLTP and the MELLLA minimum flow point. [[

]] CPPU allows plants to increase their operating thermal power but does not allow an increase in control rod line. [[

]] The conclusion of Reference 29 and the associated NRC SER that the analyzed operator actions effectively mitigate an ATWS instability event are applicable to the operating conditions expected for CPPU at VYNPS.

[[

]]

Table 9-1
LOCA Radiological Consequences

Location	Current	Limit [1]	CPPU	Limit [2]
Exclusion Area				
Whole Body Dose	0.043	≤ 25	N/A	N/A
Thyroid Dose	94	≤ 300	N/A	N/A
TEDE Dose	N/A	N/A	3.14	≤ 25
Low Population Zone				
Whole Body Dose	0.28	≤ 25	N/A	N/A
Thyroid Dose	8.4	≤ 300	N/A	N/A
TEDE Dose	N/A	N/A	0.52	≤ 25
Control Room				
Whole Body Dose	0.003	≤ 5	N/A	N/A
Thyroid Dose	20.2	≤ 30	N/A	N/A
Beta Dose	N/A	N/A	N/A	N/A
TEDE Dose	N/A	N/A	3.40	≤ 5

Notes:

1. 10 CFR 100 limit (rem)
2. 10 CFR 50.67 limit (rem TEDE)

Table 9-2
FHA Radiological Consequences

Location	Current	Limit [1.]	CPPU	Limit [2.]
Exclusion Area				
Whole Body Dose	0.027	≤ 25	N/A	N/A
Thyroid Dose	32	≤ 300	N/A	N/A
TEDE Dose	N/A	N/A	0.472	≤ 6.30
Low Population Zone				
Whole Body Dose	0.00084	≤ 25	N/A	N/A
Thyroid Dose	3.4	≤ 300	N/A	N/A
TEDE Dose	N/A	N/A	< 0.472	≤ 6.30
Control Room				
Whole Body Dose	N/A	N/A	N/A	N/A
TEDE Dose	N/A	N/A	0.153	≤ 5

Notes:

1. 10 CFR 100 limit (rem)
2. 10 CFR 50.67 limit (rem TEDE)

Table 9-3
CRDA Radiological Consequences

Location	Current	Limit [1.]	CPPU	Limit [2.]
Exclusion Area				
Whole Body Dose	0.015	≤ 25	N/A	N/A
Thyroid Dose	3.0	≤ 300	N/A	N/A
Beta Dose	0.023	≤ 300	N/A	N/A
TEDE Dose	N/A	N/A	0.38	≤ 6.30
Low Population Zone				
Whole Body Dose	0.0074	≤ 25	N/A	N/A
Thyroid Dose	1.8	≤ 300	N/A	N/A
Beta Dose	0.012	≤ 300	N/A	N/A
TEDE Dose	N/A	N/A	0.081	≤ 6.30
Control Room				
Whole Body Dose	0.0097	≤ 5	N/A	N/A
Thyroid Dose	28	≤ 30	N/A	N/A
Beta Dose	0.37	≤ 30	N/A	N/A
TEDE Dose	N/A	N/A	0.40	≤ 5

Notes:

1. 10 CFR 100 limit (rem)
2. 10 CFR 50.67 limit (rem TEDE)

Table 9-4
VYNPS Key Inputs for ATWS Analysis

Input Variable	CLTP	CPPU
Reactor power (MWt)	1593	1912
Reactor dome pressure (psia)	1025	1025
Each SRV capacity at 1080 psig (lbm/hr)	800,000	800,000
Each SSV capacity at 1240 psig (lbm/hr)	932,500	932,500
SRV / SSV Configuration	4 / 3	4 / 3
High pressure ATWS-RPT (psig)	1150	1150
Number of SRVs OOS	1	1

Table 9-5
VYNPS Results of ATWS Analysis [1.]

Acceptance Criteria	CLTP	CPPU
Peak vessel bottom pressure (psig)	1367	1490
Peak suppression pool temperature (°F)	183	190
Peak containment pressure (psig)	11.1	12.5

Note:

1. Cladding temperature and oxidation remain below their 10 CFR 50.46 limits.

10. OTHER EVALUATIONS

10.1 HIGH ENERGY LINE BREAK

HELBs are evaluated for their effects on equipment qualification. The topics addressed in this evaluation are:

Topic	CPPU Disposition	VYNPS Result
10.1.1 Steam lines	[[	
10.1.2 Liquid lines		]]

[[

]] The results of the VYNPS evaluation of HELBs are provided in Table 10-1.

10.1.1 Steam Line Breaks

The steam line HELBs in the VYNPS licensing basis were evaluated for CPPU.

Main Steam Line Breaks

CPPU has no effect on MSLBs because steam conditions at the postulated break locations are unchanged. CPPU has no effect on the steam pressure or enthalpy at the postulated break locations. Therefore, CPPU has no effect on the mass and energy releases from an HELB in a main steam line.

HPCI Steam Line Breaks

Because there is no increase in the reactor dome pressure relative to the CLTP analysis, the mass flow rates for HPCI steam line breaks do not increase. Therefore, the CLTP analysis of the HPCI steam line breaks is bounding for CPPU conditions.

RCIC Steam Line Breaks

Because there is no increase in the reactor dome pressure relative to the CLTP analysis, the mass flow rates for RCIC steam line breaks do not increase. Therefore, the CLTP analysis of the RCIC steam line breaks is bounding for CPPU conditions.

[[

]]

10.1.2 Liquid Line Breaks

Operation at CPPU conditions requires an increase in the MS and FW flows, which results in a slight increase in downcomer subcooling. This increase in subcooling may lead to increased break flow rates for liquid line breaks. Only the mass and energy releases for HELBs in the RWCU and FW systems may be affected by CPPU and were re-evaluated at CPPU conditions.

RWCU, combined RWCU and FW, and combined FW and MSL breaks were re-analyzed at CPPU conditions. The RWCU and combined RWCU and FW HELBs are slightly affected by changes in initial pressure and temperature. The effects on pressure and temperature from these HELBs in the steam tunnel and other regions within the Reactor Building were analyzed. The effect of CPPU on mass release, pressure, and temperature for various liquid line breaks and locations is summarized on Table 10-1. The effect on equipment qualification from these liquid line HELBs is discussed in Section 10.3.

Pipe Whip and Jet Impingement

Pipe whip and jet impingement loads resulting from high energy pipe breaks are directly proportional to system pressure. Because CPPU conditions do not result in an increase in pressure considered in the high-energy piping evaluations, there is no increased pipe whip or jet impingement loads on HELB targets or pipe whip restraints. The pipe stress evaluations of high energy piping systems at CPPU conditions did not result in the identification of any new pipe break locations. A review of the FW system pipe stress calculations determined that FW temperature increases associated with CPPU conditions do not result in any new postulated pipe break locations. In summary, CPPU conditions do not result in new HELB locations, nor affect existing HELB evaluations of pipe whip restraints and jet targets.

Internal Flooding from Feedwater Line Break

The flooding is dependent upon the maximum water levels in the hotwells and not CPPU reactor vessel conditions. FW system changes have been evaluated and the flooding rate from a FW line break is acceptable. Because the water level in the hotwells, the existing draining systems, and existing flood barriers are not changing, the existing FW break flooding analysis is valid for the CPPU condition.

10.2 MODERATE ENERGY LINE BREAK

A Moderate Energy Line Break (MELB) is not within the VYNPS licensing basis [[ ]] Therefore, MELB is not applicable to VYNPS for the CPPU.

10.3 ENVIRONMENTAL QUALIFICATION

Safety-related components are required to be qualified for the environment in which they are required to operate. The topics addressed in this evaluation are:

Topic	CPPU Disposition	VYNPS Result
10.3.1 Electrical Equipment	[[	
10.3.2 Mechanical Equipment With Non-Metallic Components		
10.3.3 Mechanical Component Design Qualification		]]

10.3.1 Electrical Equipment

The safety-related electrical equipment was reviewed to assure the existing qualification for the normal and accident conditions expected in the area where the devices are located remain adequate. Table 10-2 provides a listing of the EQ effects and parameter changes associated with CPPU.

Inside Containment

EQ for safety-related electrical equipment located inside the containment is based on MSLB and/or DBA/LOCA conditions and their resultant temperature, pressure, humidity, and radiation consequences, and includes the environments expected to exist during normal plant operation. Normal temperatures are expected to increase slightly, but remain bounded by the normal temperatures used in the EQ analyses. The CPPU accident conditions are compared to CLTP accident conditions in Section 4. The accident conditions for temperature and pressure, used in the current EQ analyses, bound the CPPU conditions described in Section 4.

The current radiation levels under normal plant conditions were evaluated to increase in proportion to the increase in RTP for the eight years remaining of the operating license, resulting in 4% increase over the 40 years. The accident radiation levels increase by ≤ 17% above the levels used in the current EQ Program. The total integrated doses (normal plus accident) for CPPU conditions were determined to challenge the qualification of some equipment located inside containment. Equipment that required further evaluation included certain cable types, splices, and electrical penetrations. A qualitative evaluation, using equipment-specific radiation dose assessment, indicates that with additional analysis, the equipment should be acceptable for the CPPU conditions. The EQ documentation and radiation analyses will be revised to demonstrate qualification to CPPU conditions.

Outside Containment

Accident temperature, pressure, and humidity environments used for qualification of equipment outside containment result from an MSLB, or other HELB, whichever is limiting for each plant

area, considering the safety function for the HELB. The peak HELB temperatures at CPPU RTP, in some cases, exceed the values used for equipment qualification at CLTP conditions. The temperature peaks that are not bounded by the CLTP conditions were evaluated. The qualification of several power supplies and distribution panels was exceeded. Such components will be requalified to the CPPU conditions by crediting new qualification tests, analysis, or relocation of the equipment to a milder environment location. The accident temperature resulting from a LOCA/MSLB inside containment increased some reactor building areas due to the additional heat load from the increase in wetwell temperatures. However, the increase in long-term post-accident temperatures was evaluated and determined not to adversely affect the qualification of safety-related electrical equipment.

The normal temperature, pressure, and humidity conditions in the reactor building do not change as a result of CPPU, except that the normal steam tunnel temperature is expected to increase slightly due to increased FW temperature. The current radiation levels under normal plant conditions were conservatively evaluated to increase in proportion to the increase in RTP for the remaining eight years of operation. The outside containment accident radiation levels increase by ≤ 17% above the levels used in the current EQ Program. The total integrated doses (normal plus accident) for CPPU conditions were evaluated. There were several types of equipment located outside of containment that were adversely affected by the radiation dose increase. A qualitative evaluation, using equipment specific radiation dose assessment indicates that with additional analysis, the equipment should be acceptable for the CPPU conditions. These components will require additional evaluation prior to CPPU implementation. The EQ documentation and radiation analyses will be revised to demonstrate qualification to CPPU conditions.

10.3.2 Mechanical Equipment With Non-Metallic Components

The temperatures, accident radiation level, and the normal radiation level increase slightly due to CPPU as discussed in Section 10.3.1. Although the VYNPS EQ Program does not specifically address mechanical equipment with non-metallic components, the VYNPS design control program ensures that non-metallic components (e.g., seals, gaskets, lubricants, diaphragms) are properly specified and procured for the environment in which they are intended to function.

10.3.3 Mechanical Component Design Qualification

The mechanical design of equipment/components (e.g., pumps, heat exchangers) in certain systems is affected by operation at CPPU due to increased temperatures and flows. The design qualification of mechanical components was evaluated relative to the revised operating conditions and determined to be adequate for CPPU operation.

The effects of increased fluid induced loads on safety-related components are described in Sections 3 and 4.1. Increased nozzle loads and component support loads due to the revised operating conditions were evaluated within the piping assessments in Section 3. These increased loads are insignificant, and become negligible (i.e., remain bounded) when combined with the

governing dynamic loads. Therefore, the mechanical components and component supports are adequately designed for CPPU conditions.

10.4 TESTING

Testing is required for the initial power ascension following the implementation of CPPU. The topics addressed in this section are:

Topic	CPPU Disposition	VYNPS Result
Plant/Component Testing	[[	]]
Large Transient Testing	[[ ]]	See separate attachment to LAR

Based on the analyses and GE BWR experience with uprated plants, a standard set of tests has been established for the initial power ascension steps of CPPU. These tests, which supplement the normal Technical Specification testing requirements, are as follows:

- Testing will be done in accordance with the Technical Specifications Surveillance Requirements on instrumentation that is re-calibrated for CPPU conditions. Overlap between the IRM and APRM will be assured.
- Steady-state data will be taken at points from 90% up to the 100% of the pre-CPPU RTP, so that system performance parameters can be projected for CPPU power before the pre-CPPU RTP is exceeded.
- CPPU power increases above the 100% pre-CPPU RTP will be made along an established flow control/rod line in increments of equal to or less than 5% power. Steady-state operating data, including fuel thermal margin, will be taken and evaluated at each step. Routine measurements of reactor and system pressures, flows, and vibration will be evaluated from each measurement point, prior to the next power increment. Radiation measurements will be made at selected power levels to ensure the protection of personnel.
- Control system tests will be performed for the reactor feedwater/reactor water level controls and pressure controls. These operational tests will be made at the appropriate plant conditions for that test at each of the power increments, to show acceptable adjustments and operational capability.
- Testing will be done to confirm the power level near the turbine first-stage scram bypass setpoint.

The same performance criteria will be used as in the original power ascension tests, unless they have been replaced by updated criteria since the initial test program. [[

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VYNPS does not plan to perform large transient testing as part of CPPU implementation. The justification for not performing large transient testing is provided as a stand-alone attachment to the CPPU LAR.

10.5 INDIVIDUAL PLANT EVALUATION

PRAs are performed to evaluate the risk of plant operation. The topics considered in this section are:

Topic	CPPU Disposition	VYNPS Result
10.5.1 Initiating Event Frequency	[[	
10.5.2 Component Reliability		
10.5.3 Operator Response		
10.5.4 Success Criteria		
10.5.5 External Events		
10.5.6 Shutdown Risk		
10.5.7 PRA Quality		]]

The VYNPS Probabilistic Safety Analyses (PSA) modeling is highly detailed, including a wide variety of initiating events, modeled systems, operator actions, and common cause events.

<u>Background</u>

The VYNPS PSA is a state-of-the-technology tool developed consistent with current PSA methods and approaches. The VYNPS PSA model was developed and quantified using the RISKMAN software. The VYNPS PSA is derived based on realistic assessments of system capability over the 24-hour mission time of the PSA analysis. Therefore, PSA success criteria may be different than the design basis assumptions used for licensing VYNPS. This risk assessment examines the risk profile changes from this realistic perspective to identify changes in the risk profile on a best estimate basis that may result from postulated accidents, including severe accidents.

<u>Scope of CPPU Risk Evaluation</u>

The scope of the risk assessment for the VYNPS CPPU addresses the following plant risk contributors:

- Level 1 Internal Events At-Power (Core Damage Frequency (CDF))

- Level 2 Internal Events At-Power (Large Early Release Frequency (LERF))
- External Events At-Power
  - Seismic Events
  - Internal Fires
  - Other External Events
- Shutdown Assessment

Risk effects due to internal events were assessed using the VYNPS Level 1 and Level 2 PSA Model of Record (Reference 30). External events were evaluated using the analyses of the VYNPS Individual Plant Examination of External Events (IPEEE) Submittal (Reference 31). The effect on shutdown risk contributions were evaluated on a qualitative basis.

All commitments resulting from the VYNPS IPE and the IPEEE Programs have been resolved.

The identification of PSA elements evaluated in the risk assessment was derived from the NEI PRA Peer Review Guidelines. Each of these major risk elements were examined:

- Initiating Events
- Systemic/Functional Success Criteria
- Accident Sequence Modeling
- System Modeling
- Failure Data
- Human Reliability Analysis
- Structural Evaluations
- Quantification
- Containment Response (Level 2)

In addition, shutdown risk and external events were also investigated.

Effect on Internal Events Plant Risk Profile

Based on the model effect discussed previously, the CPPU is estimated to increase the VYNPS internal events PSA CDF from the CLTP value of 7.77E-6/yr to 8.10E-6/yr, which is an increase of 3.3E-7/yr (4.2%). The composition and comparative distribution of the CPPU results remain unchanged with respect to the CLTP VYNPS PSA (see Table 10-3).

The at-power internal events LERF increased from the CLTP value of 2.23E-6/yr to 2.34E-6/yr for CPPU, which is an increase of 1.1E-7/yr (4.9%).

Quantitative Sensitivity Cases

In addition to the base (best estimate) CPPU quantification, six quantitative sensitivity cases were performed. These cases are summarized in Table 10-4.

Overview of CPPU Modifications

Hardware Modifications

The hardware modifications to be implemented as part of the CPPU and considered in the CPPU PSA evaluation are as follows:

Mechanical

- Replacement of the high pressure (HP) turbine rotor (and modification of the turbine controls)
- Replacement of the main generator hydrogen cooling system heat exchangers
- BOP and NSSS pipe support modifications
- Modification to RHRSW motor cooling piping
- Replacement of HP FW heaters in both trains
- Modifications to the IPB cooling system (additional cooling capacity necessary to support new power level)
- Increased condensate demineralizer flow capacity
- Stake main condenser tubes
- RPV steam dryer modification for structural integrity

Electrical

- Rewind of main turbine generator
- Upgrade generator disconnect switch
- Addition of recirculation pump runback logic
- Various instrumentation and controls component upgrades/replacements (e.g., FW Level Control (FWLC))

In addition, the modification to add an additional SSV for ARTS/MELLLA and transition to the GE14 fuel type initiated in Cycle-23 were considered in this evaluation.

Procedural Changes

Adjustments to the VYNPS Emergency Operating Procedures (EOPs) / Severe Accident Management Guidelines (SAMGs) will be made to be consistent with CPPU operating conditions. In almost all respects, the EOPs/SAMGs are expected to remain unchanged because

they are symptom-based; however, certain parameter thresholds and graphs are dependent upon power and decay heat levels and will require slight modifications.

Based on the CPPU evaluations, EOP variables that play a role in the PSA and which may require adjustment for the CPPU include:

- Boron Injection Initiation Temperature (BIIT)
- Heat Capacity Temperature Limit (HCTL)
- Pressure Suppression Pressure Limit (PCPL)

These variables may require adjustment to reflect the change in power level, but will not be adjusted in a manner that involves a change in accident mitigation philosophy. The HCTL and PCPL relate to long-term scenarios. Any change in the scenario timings associated with CPPU curve changes will be minor (e.g., changes on the order of 10-15 minutes over accident times greater than three hours) and would not significantly affect the Human Error Probabilities (HEPs) in the PSA.

Any CPPU-related changes to the VYNPS EOPs or SAMGs are considered minor perturbations. Therefore, the EOP/SAMG changes as a result of the CPPU will not influence the risk profile.

Setpoint Changes

The RPV operating pressure and temperature are not being changed as part of the CPPU. Potential setpoint changes for the CPPU include:

- Turbine overspeed
- TFSP steam scram bypass
- MSL High Flow Isolation

CPPU does not require changes to the following setpoints:

- RPT/ATWS high dome pressure
- RPV level trips/actuations
- MSL low pressure isolation
- SSV/SRV setpoints

Plant Operating Conditions

The key plant operational modifications to be made in support of the CPPU are:

- Increase in RTP from 1593 to 1912 MWt
- FW/Condensate flow (and steam flow) rates increase by approximately 20% over CLTP
- Operation of all three RFPs

RPV pressure remains unchanged for the CPPU.

10.5.1 Initiating Event Frequency

The evaluation of the plant and procedural changes indicates no new initiators or increased frequencies of existing initiators are anticipated to result from CPPU.

The VYNPS PSA program encompasses an effectively exhaustive list of hazards and accident types (i.e., from simple non-isolation transients to ATWS scenarios to internal fires to hurricanes to toxic releases to draindown events during refueling activities, and numerous others). Extensive and unique changes to the plant would have to be implemented to result in new previously unidentified accidents. This is not the case for the VYNPS CPPU.

The VYNPS PSA initiating events can be categorized into the following:

- Transients
- LOOP
- LOCAs
- Support System Failures
- Internal Floods
- External Events

Transients

The transients are the only initiating events that may be realistically postulated to experience an effect to the calculated initiating event frequencies.

An increase in the TT initiator frequency can be postulated and calculated due to the increase in the required number of RFPs (i.e., all three RFPs required for power operation for the CPPU condition). This potential increase in TT initiator frequency is estimated in the VYNPS CPPU risk assessment as follows:

- Based on Section 4.5 of NUREG/CR-5750, 8% of the U.S. BWR industry TT frequency is due to partial LOFW events.
- The CLTP PSA TT frequency is 0.55/yr.
- The CLTP configuration requires two running RFPs, and the CPPU configuration requires three running RFPs.
- The third RFP requirement adds the following potential failure set: "FW pump "C" fails to run" x "FW control failure." Similar failure sets for FW pump A and B are already implicitly represented in the CLTP TT frequency.

- Conservatively assuming that the partial LOFW contribution to the TT frequency is comprised of just these three failure sets, the CPPU TT frequency may be calculated as follows:

  $$[0.55 \times (1\text{-}0.08)] + [0.55 \times 0.08 \times (3/2)] = 5.7\text{E-}1$$

The requirement for the additional running RFP is reasonably modeled here as affecting the TT frequency (which encompasses a reactor trip, the likely direct effect of an RFP trip). Total LOFW is a separate initiator; however; Total LOFW would be negligibly affected by an additional running RFP. Total LOFW is overwhelmingly dominated by other issues (e.g., FW regulating valve closure).

LOOP

No change in the LOOP initiating event frequency is expected. Currently, VYNPS has certain operational configurational conditions that require power reductions to maintain grid stability. The same or similar conditions and operations will exist for the CPPU. Therefore, CPPU is not expected to have any effect on the grid related LOOP initiating event frequency.

LOCAs

No changes to RPV operating pressure, inspection frequencies, or primary water chemistry are planned in support of the CPPU. As such, no effect on LOCA frequencies due to the CPPU can be postulated. However, a sensitivity case was analyzed that <u>doubles</u> the Large LOCA initiator frequency.

Support System Failures

No significant changes to support systems (e.g., Instrument Air, Service Water) are planned in support of the CPPU. As such, no effect on support system initiating event frequencies due to the CPPU can be postulated.

Internal Floods

No changes to pipe inspection scopes or frequencies are planned in support of the CPPU. As such, no effect on internal flooding initiator frequencies due to the CPPU can be postulated.

External Events

The frequency of external event initiators (e.g., seismic events, extreme winds, fires) is not linked to reactor power or operation. As such, no effect on external event initiator frequencies due to the CPPU can be postulated.

10.5.2 Component Reliability

The majority of the hardware changes in support of the CPPU may be characterized as either:

- Replacement of components with enhanced like components
- Upgrade of existing components

Although equipment reliability as reflected in failure rates can be theoretically postulated to behave as a "bathtub" curve (i.e., the beginning and end of life phases being associated with higher failure rates than the steady-state period), no significant effect on the long-term average of initiating event frequencies, or equipment reliability during the 24 hour PSA mission time due to the replacement/modification of plant components is anticipated, nor is such a quantification supportable at this time. If any component degradation were to occur as a result of CPPU implementation, existing plant monitoring programs would address any such issues.

10.5.3 Operator Response

The VYNPS risk profile, like other plants, is dependent on the operating crew actions for successful accident mitigation. The success of these actions is in turn dependent on a number of performance shaping factors. The performance-shaping factor that is principally influenced by CPPU is the time available within which to detect, diagnose, and perform required actions. The higher power level results in reduced times available for some actions. To quantify the potential effect of this performance-shaping factor, deterministic thermal hydraulic calculations using the MAAP computer code were used.

Not all operator actions in the VYNPS PSA have a significant effect on the results. To minimize the resources required to requantify all operator actions in the PSA due to the CPPU, a screening process was first performed to identify those operator actions that have an effect on the PSA results. This is consistent with past CPPU risk assessments and is reasonable.

The screening process was performed against the following criteria:

1. F-V (with respect to CDF) importance measure ≥ 5E-3
2. RAW (with respect to CDF) importance measure ≥ 2.0
3. F-V (with respect to LERF) importance measure ≥ 5E-3
4. RAW (with respect to LERF) importance measure ≥ 2.0
5. Time critical (≤ 30 min. available) action

If any of the above criteria are met for an operator action the action is maintained for explicit consideration in the CPPU risk assessment. Potential HEP changes for operator actions screened out from explicit assessment in this CPPU risk assessment will not have an effect on the quantitative results. Of all the actions screened from further analysis, only three actions when assumed failed with an HEP of 1.0 would result in an increase in CDF by ≥ 1E-6 or LERF by ≥1E-7. However,

each of these three screened actions has a long (≥ 1 hour) allowable response time such that the HEPs would not be affected by the CPPU.

Over three dozen operator actions were identified for explicit consideration. MAAP calculations for the VYNPS CLTP and CPPU configurations were performed to determine changes in allowable operator action timings. The HEPs were then re-calculated using the same Human Reliability Analysis (HRA) methods used in the VYNPS PRA. Refer to Table 10-5 for a summary of the changes in operator action timings and associated HEPs.

The risk importance measures of these actions change slightly for the CPPU but do not result in changing their relative significance to the VYNPS risk profile. Using the $FV_{CDF} \geq$ 5E-3 and $RAW_{CDF} \geq 2.0$ as the criteria for risk significance of the operator actions, only a single operator action moved up past this risk significance test threshold. This action, QOP003FL, has a RAW of just barely under 2.0 for the CLTP case, and increases to a RAW of approximately 3 for the CPPU quantification. This does not change the relative significance of this action to the plant risk profile. As such, no new risk significant operator actions resulted from this analysis.

A new operator action will be incorporated into plant procedures to satisfy certain aspects of Appendix R and SBO evaluations at CPPU. This action is to close a normally open torus vent line. Closure of this line (which is a simple action performed using Control Room switches) allows credit for containment overpressure to maintain ECCS NPSH. However, this action has no direct applicability to the PSA as the PSA credits torus cooling (in which case the action is moot) whereas the Appendix R and SBO evaluations requiring this action do not.

The CPPU PSA includes the implementation of an operational enhancement that will provide for automatic recirculation run-back given a single RFP trip. This operational enhancement can be considered as an automation of an operator action. As this feature did not exist previously, it is not in the CLTP VYNPS PSA. It is addressed in this risk assessment in the adjustments to the TT initiator frequency.

The CPPU does not affect operator workarounds, as VYNPS does not have any current operator workarounds.

No significant changes are to be made to the Control Room for the CPPU that would affect the VYNPS PSA HRA. Changes to be made to the Control Room displays for the CPPU are:

- MSL flow indicators replaced with digital units
- FW flow indicators replaced with digital units
- Steam/FW Flow recorder re-scaled
- Condensate Flow recorder re-scaled

None of these Control Room display changes affect in any way the HRA for the VYNPS PSA.

10.5.4 Success Criteria

The success criteria for the VYNPS PSA are derived based on realistic evaluations of system capability over the 24-hour mission time of the PSA analysis. Therefore, these success criteria may be different than the design basis assumptions used for licensing VYNPS. The CPPU PSA anlaysis examines the risk profile changes caused by CPPU from a realistic perspective to identify changes in the risk profile that may result from severe accidents on a best estimate basis. The only success criteria effect caused by the CPPU is the number of SRVs/SSVs required for initial RPV overpressure control during an isolation ATWS scenario. The pre-ARTS/MELLLA plant configuration included four SRVs and two SSVs. The CPPU plant configuration, which includes ARTS/MELLLA, has an additional SSV. The additional SSV provides additional relief capacity for the limiting ATWS transient. The CPPU configuration is more than adequate with one SRV OOS. The CPPU risk assessment includes the effect of the additional SSV on the pressure relief success criteria during an ATWS scenario and the increased challenge presented by the CPPU plant configuration (e.g., power level, rod line). This success criteria change is addressed in the VYNPS CPPU risk assessment.

No changes in success criteria were identified with regard to the VYNPS Level 2 (containment evaluation) PRA. The slight changes in accident progression timing and decay heat load have only minor or negligible impacts on Level 2 PSA safety functions, such as containment isolation, ex-vessel debris coolability, and challenges to the ultimate containment strength.

<u>Stuck-Open Relief Valve Probability</u>

A model effect that may be classified as a success criteria issue is that related to the Stuck-Open Relief Valve (SORV) probability.

The SRV setpoints have not been changed as a result of CPPU. Given the power increase of the CPPU, one may postulate that the probability of an SORV given a transient initiator would increase due to an increase in the number of SRV cycles.

The SORV probability used in the VYNPS PSA is 1.08E-2. The CPPU PSA base probability for an SORV may be modified using different approaches to consider the effect of a postulated increase in valve cycles. The following three approaches were considered:

1. The upper bound approach would be to increase the SORV probability by a factor equal to the increase in reactor power (i.e., a factor of 1.2 for the VYNPS CPPU of 120% of CLTP). This approach assumes that the SORV probability is linearly related to the number of SRV cycles, and that the number of cycles is linearly related to the reactor power increase.
2. A less conservative approach to the upper bound approach would be to assume that the SORV probability is linearly related to the number of SRV cycles. However, the number of cycles is not necessarily directly related to the reactor power increase. In this case, the postulated increase in SRV cycles due to the CPPU would be determined by thermal hydraulic calculations (e.g., MAAP runs).

3. The lower bound approach would be to assume that the SORV probability is dominated by the initial cycle and that subsequent cycles have a much lower failure rate. In this approach, the CLTP SORV probability could be assumed to be insignificantly changed by a postulated increase in the number of SRV cycles.

Approach #2 was used to modify the VYNPS PSA SORV probability for CPPU. The increase in the number of SRV cycles during accident response was estimated by comparing the results of MAAP runs for isolation transient scenarios performed in support of the CPPU risk assessment. MAAP cases VYEPU2 and VYEPU2x indicate that the number of SRV cycles in the first couple hours of the accident progression increases by 13% for CPPU versus CLTP. Similarly, MAAP cases VYEPU5b and VYEPU5bx indicate that the number of cycles increases 15%.

Using this information, the VYNPS PSA CLTP SORV probability given a transient initiator of 1.08E-2 is increased 15% (to 1.24E-2) to represent the CPPU configuration.

Accident Sequence Modeling

The CPPU does not change the plant configuration and operation in a manner such that new accident sequences or changes to existing accident scenario progressions result. A slight exception is the reduction in available accident progression timing for some scenarios and the associated effect on operator action HEPs (this aspect is addressed in the Human Reliability Analysis section).

System Modeling

The VYNPS plant changes associated with the CPPU do not result in the need to change any system modeling in support of this risk assessment.

Note that the increase in the number of normally running RFPs affects the initiating event frequencies, and not the post-scram mitigative and support system models of the PRA.

Level 2 PSA Analysis

Given the minor change in Level 1 CDF results, minor changes in the Level 2 PSA release frequencies result. Such changes are directly attributable to the change in the TT initiating event frequency and the minor changes in short term accident sequence timing and the effect on HEPs. The accident sequence modeling in the Level 2 PSA is not affected by the CPPU.

Fission product inventory in the reactor core is higher as a result of the increase in power due to the CPPU. The increase in fission product inventory results in an increase in the total radionuclides available for release given a severe accident. The total activity available for release is approximately 20% higher. However, this does not effect the definition or quantification of the LERF risk measure used in RG 1.174, and is the basis for this risk assessment. The VYNPS PSA release categories are defined based on the percentage (as a

function of EOC inventories) of Cesium Iodide released to the environment. This is consistent with most industry PSAs.

10.5.5 External Events

Internal Fires

The VYNPS plant risk due to internal fires was evaluated in 1998 as part of the VYNPS IPEEE Submittal. The EPRI Fire Induced Vulnerability Evaluation (FIVE) Methodology (Reference 32) and the Fire PRA Implementation Guide screening approaches and data were used to perform the VYNPS IPEEE fire PSA study.

Consistent with the FIVE Methodology and the requests of the NRC IPEEE Program, the VYNPS IPEEE fire PSA is an analysis that identifies the most risk significant fire areas in the plant using a screening process and by calculating conservative core damage frequencies for fire scenarios. As such, the accident sequence frequencies calculated for the VYNPS fire PSA are not a best estimate calculation of plant fire risk and are not acceptable for integration with the best estimate VYNPS internal events PSA results for comparison with RG 1.174 acceptance guidelines. As such, a qualitative evaluation on the VYNPS fire risk profile due to the CPPU was performed based on review of the VYNPS IPEEE fire PSA results. This estimate is performed as follows:

- The dominant fire scenarios from the VYNPS IPEEE fire analysis are used to represent the VYNPS fire risk profile.
- The quantitative results (i.e., RISKMAN code sequences results) of these fire scenarios are reviewed to breakdown each fire scenario into three accident classes:
  - Loss of coolant makeup accident (i.e., Class I and III core damage accidents)
  - Loss of containment heat removal accidents (i.e., Class II core damage accidents)
  - ATWS accidents (i.e., Class IV core damage accidents)
- The results of the CPPU base case quantification (in terms of percentage CDF increase as a function of accident class) are applied to the VYNPS fire scenarios.
- As the VYNPS fire scenarios are significantly dominated by fire-induced equipment failures (typical of fire PSAs), the percentage CDF increases manifested in the internal events due to the CPPU are not directly applicable to the fire-induced core damage accidents. The fire-induced accidents are less affected by changes in operator actions timings than the internal events. The internal events percentage CDF changes are reduced by a factor of 0.5 before application to the fire scenarios. This is considered reasonable.

The fire effect calculation estimate is summarized in Table 10-6. As can be seen from Table 10-6, it is estimated that the VYNPS fire PSA CDF would increase by approximately 1.5% due to the CPPU.

Seismic Risk

The VYNPS seismic risk analysis was performed as part of the IPEEE. VYNPS performed a Seismic Margins Assessment (SMA) following the guidance of NUREG-1407 and EPRI NP-6041. The SMA is a deterministic evaluation process that does not calculate risk on a probabilistic basis. No CDF sequences were quantified as part of the seismic risk evaluation.

The conclusions of the VYNPS IPEEE seismic analysis are as follows:

> For VYNPS, the SMA identified that the lowest High Confidence of Low Probability of Failure (HCLPF) components in the selected primary and alternate safe shutdown paths are the CST with a HCLPF of 0.25g and the Diesel Fuel Oil Storage Tank (FOST) with a HCLPF of 0.29g. The HCLPF for all other components in the safe shutdown paths meet or exceed the 0.3g review level earthquake. These values represent significant margin to the design basis 0.14g earthquake.

Based on a review of the VYNPS IPEEE and the key general conclusions identified earlier in this assessment, the conclusions of the SMA are considered to be unaffected by the CPPU. The CPPU has little or no effect on the seismic qualifications of the Systems, Structures and Components (SSCs). Specifically, CPPU results in additional thermal energy stored in the RPV, but the additional blowdown loads on the RPV and containment given a coincident seismic event, do not alter the results of the SMA.

The decrease in time available for operator actions, and the associated increases in calculated HEPs, are considered to have an insignificant effect on seismic-induced risk. Industry BWR seismic PSAs have typically shown (e.g., Peach Bottom NUREG-1150 study; Limerick Generating Station Severe Accident Risk Assessment; NUREG/CR-4448) that seismic risk is overwhelmingly dominated by seismic induced equipment and structural failures.

Based on the above discussion, it is concluded that the percentage increase in the VYNPS seismic risk due to the CPPU is much less than that calculated for internal events.

Other External Events Risk

In addition to internal fires and seismic events, the VYNPS IPEEE Submittal analyzed a variety of other external hazards:

- High Winds/Tornadoes
- External Floods
- Transportation and Nearby Facility Accidents
- Other External Hazards

The VYNPS IPEEE analysis of high winds, tornadoes, external floods, transportation accidents, nearby facility accidents, and other external hazards was accomplished by reviewing the plant

environs against regulatory requirements regarding these hazards. Based on this review, it was concluded that VYNPS meets the applicable NRC Standard Review Plan requirements and therefore has an acceptably low risk with respect to these hazards.

Note that the VYNPS IPEEE also analyzed internal flooding scenarios. In addition, internal flooding scenarios were incorporated into the VYNPS CLTP PSA internal events model. The effect on internal flooding accident sequences is addressed quantitatively in this CPPU risk assessment as part of the internal events risk impact.

10.5.6 Shutdown Risk

The effect of the CPPU on shutdown risk is similar to the effect on the at-power Level 1 PSA. Based on the insights of the at-power PSA effect assessment, the areas of review appropriate to shutdown risk are the following:

- Initiating Events
- Success Criteria
- Human Reliability Analysis

The following qualitative discussion applies to the shutdown conditions of Hot Shutdown (Mode 3), Cold Shutdown (Mode 4), and Refueling (Mode 5). The CPPU risk effect during the transitional periods such as at-power (Mode 1) to Hot Shutdown and Startup (Mode 2) to at-power are judged to be subsumed by the at-power Level 1 PSA.

<u>Shutdown Initiating Events</u>

Shutdown initiating events include the following major categories:

- Loss of RCS Inventory
  - Inadvertent Draindown
  - LOCAs
- Loss of DHR (includes LOOP)

No new initiating events or increased potential for initiating events during shutdown (e.g., loss of DHR train) can be postulated due to CPPU.

<u>Shutdown Success Criteria</u>

The effect of the CPPU on the success criteria during shutdown is similar to the Level 1 PSA. The increased power level decreases the time to boildown. However, because the reactor is already shutdown, the boildown times are much longer compared to the at-power PSA. The boildown to TAF time is approximately 2 hours at 2 hours after shutdown (e.g., time of Hot Shutdown) and approximately 4-6 hours at 12-24 hours after shutdown (e.g., time of Cold

Shutdown). The decrease in the boildown time for the CPPU is small because of the lower decay heat level relative to at-power conditions.

The increased decay heat loads associated with the CPPU affects the time when low capacity DHR systems can be considered successful alternate DHR systems. CPPU delays the time after shutdown when low capacity DHR systems may be used as an alternative to SDC. However, shutdown risk is dominated during the early time frame soon after shutdown when the decay heat level is high and, in this time frame, low capacity DHR alternatives are already not viable DHR systems.

Other success criteria are marginally affected by the CPPU. The CPPU has a minor effect on shutdown RPV inventory makeup during loss of DHR scenarios in shutdown because of the low decay heat level. The heat load to the suppression pool during loss of DHR scenarios in shutdown (i.e., during shutdown phases with the RPV intact) is also lower because of the low decay heat level such that the margins for suppression pool cooling capacity are adequate for CPPU.

The CPPU effect on the success criteria for blowdown loads, RPV overpressure margin, and SRV actuation is estimated to be negligible because of the low RPV pressure and low decay heat level during shutdown.

Shutdown HRA Effect

Similar to the at-power Level 1 PSA, the decreased boildown time due to the CPPU decreases the time available for operator actions. The significant, time critical operator actions affected in the at-power Level 1 PSA are related to RPV depressurization, SLC injection, and SLC level control. These operator actions do not directly apply to shutdown conditions because the RPV is at low pressure and the reactor is subcritical. The risk significant operator actions during shutdown conditions include recovering a failed DHR system or initiating alternate DHR systems. However, the longer boildown times during shutdown results in the CPPU having a minor effect on the shutdown HEPs associated with recovering or initiating DHR systems.

Thermal hydraulic calculations performed for this risk assessment for the VYNPS water volumes during shutdown conditions show that the times available to perform loss of DHR response actions during shutdown is many hours. The reductions in these times due to the CPPU is shown in the range of 10 to 15% (depending on time after shutdown and water level configuration). Such small changes in already lengthy operator action response times result in negligible changes in HEPs. Changes in AC power recovery probabilities were also investigated.

Assuming a typical 30-day refueling outage, the effect on shutdown risk due to these timing reductions was estimated as an approximate 2% increase in shutdown CDF.

Shutdown Risk Summary

Based on a review of the potential effect on initiating events, success criteria, and HRA, CPPU is considered to have an insignificant effect (ΔCDF of roughly 2%) on shutdown risk.

VYNPS Outage Risk Management Process

VYNPS uses a computerized risk monitor and site-specific matrices as tools for controlling outage risk. The effect of the outage activities upon key safety functions is assessed as follows:

- Identify key safety functions affected by the SSC planned for removal from service.
- Consider the degree to which removing the SSC from service will affect the key safety functions.
- Consider degree of redundancy, duration of out-of-service condition, and appropriate compensatory measures, contingencies, or protective actions that could be taken if appropriate for the activity under consideration.

The Key Safety Function Matrices were developed from guidance provided by NUMARC 91-06 and NUMARC 93-01. The shutdown key safety functions are achieved by using systems or combinations of systems. The SSCs to be addressed by the assessment for shutdown conditions are those SSCs necessary to support the following shutdown key safety functions (from Section 4 of NUMARC 91-06):

- DHR capability
- Inventory control
- Power availability
- Reactivity control
- Containment (primary/secondary)

Managing the risk involves invoking some or all of the following elements:

- Pre-job briefs of operating and maintenance crews
- System engineering oversight
- Management oversight
- Outage management approval of the proposed activity
- Pre-staged parts and materials
- Walkdown of tagouts and maintenance activity prior to conducting the maintenance
- Mockup training
- Reduce OOS time through overtime or additional shift coverage
- Contingency plans for returning equipment to service in a timely manner if needed

- Compensatory measures to minimize initiators and/or mitigate the consequences
- Reschedule or minimize work on functionally related equipment.
- Proceduralize other success paths of the safety function affected

10.5.7 PRA Quality

The quality of the VYNPS PSA models used in performing the risk assessment for the VYNPS CPPU is manifested by the following:

- Sufficient scope and level of detail in PSA
- Active maintenance of the PSA models and inputs
- Comprehensive Critical Reviews

Scope and Level of Detail

The VYNPS PSA is of sufficient quality and scope for CPPU. The VYNPS PSA modeling is highly detailed, including a wide variety of initiating events (e.g., transients, internal floods, LOCAs inside and outside containment, support system failure initiators), modeled systems, extensive level of detail, operator actions, and common cause events.

Maintenance of Model, Inputs, Documentation

The VYNPS PSA model and documentation has been updated to reflect the current plant configuration and to reflect the accumulation of additional plant operating history and component failure data. The VYNPS CLTP PSA model at the time of this analysis is VY02 Revision 6 (Reference 30). The Level 1 and Level 2 VYNPS PSA analyses were originally developed and submitted to the NRC in December 1993 as the VYNPS IPE Submittal (Reference 33).

Critical Reviews

The VYNPS internal events received a formal industry PRA Peer Review in November 2000. All of the "A" and "B" priority comments have been addressed by VYNPS in the VYNPS CLTP PSA model as appropriate.

10.5.8 Conclusion

The key result of the PSA evaluation is that only small risk increases were calculated for both CDF and LERF. The risk increase is associated with reduced times available for certain operator actions and the assumed increase in the TT initiating event frequency.

The best estimate of the risk increase for at-power internal events due to the CPPU is a ΔCDF of 3.3E-7/yr (an increase of 4.2% over the CLTP CDF of 7.77E-6/yr). The best estimate at-power

internal events LERF increase due to the CPPU is a ΔLERF of 1.1E-7/yr (an increase of 4.9% over the CLTP LERF of 2.23E-6/yr).

Using the NRC guidelines established in RG 1.174 and the calculated results from the Level 1 and 2 PSA, the best estimate for the VYNPS CDF risk increase due to the CPPU (3.3E-7/yr) is well within Region III (i.e., changes that represent very small risk changes). The best estimate for the LERF increase (1.1E-7/yr) is in Region II, but close to the Region III criteria. Region II is identified as changes that represent small risk changes.

10.6 OPERATOR TRAINING AND HUMAN FACTORS

Some additional training is required to enable plant operation at the CPPU RTP level. The topic addressed in this section is:

Topic	CPPU Disposition	VYNPS Result
Operator training and human factors	[[	]]

For CPPU conditions, operator responses to transients, accidents, and special events are minimally affected. One operator transient response, the required action following a RFP trip, will be changed to reflect the installation of an automatic recirculation system runback. Operator response to a fire in the reactor building Appendix R event will require closing, from the Control Room, a normally open torus vent. This added task is necessary to ensure that containment overpressure is available to maintain adequate NPSH for the ECCS pumps. Also, the time available for some operator actions is reduced by small increments. Most abnormal events result in an automatic plant shutdown (scram). Some abnormal events result in automatic RCPB pressure relief, ADS actuation and/or automatic ECCS actuation (for low water level events). All events result in safety-related Systems, Structures, and Components (SSCs) remaining within their design allowables. CPPU does not change any of the automatic safety functions, except for those setpoint changes identified in Section 5.3. After the applicable automatic responses have initiated, the subsequent operator actions for plant safety (e.g., maintaining safe shutdown, core cooling, and containment cooling) do not change for CPPU.

CPPU system changes were analyzed for the effect on NSSS and BOP instrumentation. The following Control Room instrumentation modifications are required as a result of this analysis: New digital bar graph indicators with new scales will be installed for the MS flow and the FW flow indicators. A new scale and chart paper will be required for the MS flow/FW flow recorder. The condensate flow indicator will require re-scaling. These, and all CPPU modifications, will be implemented in accordance with the VYNPS design modification processes. The VYNPS design change process requires human factors review for modifications and includes impact review by operations and training personnel. Operator training for these modifications is described below.

The analog and digital inputs for the Emergency Response Facility Information System (ERFIS) including the Safety Parameter Display System (SPDS) will be reviewed to determine the effect

of CPPU. This includes required changes to monitored points, calculations, and alarm setpoints. Various changes in Emergency Operating Procedure (EOP) curves and limits, if required, will also require an update of the SPDS. Any changes required to the ERFIS computer will be completed prior to CPPU implementation.

Following a review of the CPPU modifications and identified key procedure changes, recommendations for operator training and simulator changes and a final determination of the operator training needs will be made, consistent with the VYNPS training program for selection of modifications for operator training. Any modifications required for CPPU will be evaluated for its effect on ERFIS and the SPDS and any required changes (including any new monitoring points) will be addressed as a part of the modification. Any changes made will be discussed as a part of the operator training program for CPPU.

The last training phase of each cycle includes training on design changes to be implemented during the upcoming outage. Lesson plans will be developed and operator classroom and simulator training will be performed prior to restart of the unit from the outage implementing the CPPU modifications. This training will cover plant modifications, procedure changes, startup test procedures, and other aspects of CPPU including changes to parameters, setpoints, scales, and systems. The applicable existing lesson plans will be revised to reflect changes as a result of the CPPU. Simulator training during this phase will also include training on performance of the new high pressure turbine and power ascension to current maximum power.

Operator Training for CPPU conditions will be performed on the simulator prior to operating the unit at CPPU conditions. The training phase prior to CPPU implementation will complete the recommended operator classroom and simulator training for CPPU implementation. This training will include the normal operating procedure actions required to achieve the CPPU RTP level, power ascension testing that will be performed, and comparisons of plant conditions between the current RTP level and the CPPU RTP level. The training will also include RFP trip and recirculation pump runback, the new operator action for the Appendix R fire in the reactor building, and selected transients and accidents that present the greatest change from previous power levels. Data obtained during startup testing will be incorporated into the training as needed.

Installation of the CPPU changes to the Simulator is planned prior to CPPU implementation. The simulator changes will include hardware changes for the new digital bar graph indicators for MS flow and FW flow and new scale and chart paper for the MS flow/FW flow recorder. Software updates for modeling changes due to CPPU (i.e., RFP and condensate pump performance upgrades, and HP turbine modifications), and setpoint changes will also be installed.

Simulator software modeling reflecting the reactor changes as a result of the CPPU will be implemented prior to the operator training session before the CPPU is initiated. Simulator acceptance testing will be conducted to benchmark the simulator performance based on design and engineering analysis data in accordance with ANSI/ANS 3.5-1998. Validation will be

performed in two phases. First, the simulator performance will be validated against the CPPU expected response. Second, Operating data will be collected during CPPU implementation and start-up testing. This data will be compared with simulator performance data, allowing any necessary adjustments to be made to the simulator model.

10.7 PLANT LIFE

The plant life evaluation identifies degradation mechanisms influenced by increases in fluence and flow. The topics addressed in this evaluation are:

Topic	CPPU Disposition	VYNPS Result
Irradiated Assisted Stress Corrosion Cracking	[[	
Flow Accelerated Corrosion		]]

VYNPS has a procedurally controlled program for the augmented Nondestructive Examination (NDE) of selected RPV internal components in order to ensure their continued structural integrity. The inspection techniques utilized are primarily for the detection and characterization of service-induced, surface-connected planar discontinuities, such as Intergranular Stress Corrosion Cracking (IGSCC) and Irradiation-Assisted Stress Corrosion Cracking (IASCC), in welds and in the adjacent base material. VYNPS belongs to the BWRVIP organization and implementation of the procedurally controlled program is consistent with the BWRVIP issued documents. The inspection strategies recommended by the BWRVIP consider the effects of fluence on applicable components and are based on component configuration and field experience.

Components selected for inspection include those that are identified as susceptible to in-service degradation and augmented examination is conducted for verification of structural integrity. These components have been identified through the review of NRC Inspection and Enforcement Bulletins (IEBs), BWRVIP documents, and recommendations provided by General Electric Service Information Letters (GE SILs). The inspection program provides performance frequency for NDE and associated acceptance criteria. Components inspected include the following:

- CS piping
- CS spargers
- Core shroud and core shroud support
- Jet pumps and associated components
- Top guide
- Lower plenum
- Vessel ID attachment welds
- Steam dryer
- FW spargers

Continued implementation of the current procedure program assures the prompt identification of any degradation of reactor vessel internal components experienced during CPPU operating conditions. To mitigate the potential for IGSCC and IASCC, VYNPS utilizes noble metals application. Reactor vessel water chemistry conditions are also maintained consistent with the Electric Power Research Institute (EPRI), BWRVIP, and established industry guidelines, except where technical justifications in accordance with BWRVIP-94 have been documented.

The service life of most equipment is not affected by CPPU. [[

]] The current inspection strategy for the reactor internal components is expected to be adequate to manage any potential effects of CPPU.

The VYNPS procedurally controlled Piping Flow-Accelerated Corrosion (FAC) Inspection Program uses selective component inspections to provide a measure of confidence in the condition of piping systems susceptible to FAC. These selective inspections are the basis for qualifying un-inspected components for further service. This approach is based upon program guidelines developed by the EPRI and the ASME. In addition to this long-term monitoring program, selected piping replacements have been performed to maintain suitable design margins. Where possible, FAC resistant replacement materials are used to mitigate future occurrences of FAC.

Component inspections are performed each refueling outage to monitor piping for FAC. The scope of these inspections is based on the combination of results from a CHECWORKS™ FAC predictive model, previous inspections at VYNPS, and industry experience obtained through INPO and by participation in the EPRI CHECWORKS™ Users Group (CHUG). The CHECWORKS™ FAC model is periodically updated to include previous inspection data.

Variables that influence FAC include:

- Moisture content
- Water chemistry
- Temperature
- Oxygen
- Flow path geometry and velocity
- Material composition

VYNPS has evaluated CPPU system operating conditions for changes in FAC effects on plant piping and components. Implementation of CPPU primarily affects moisture content, temperature, oxygen, and flow velocity. The magnitude of predicted FAC wear rates increase and vary throughout the BOP piping due to the increased flows, temperatures, and the moisture removal capabilities of plant equipment. Table 10-7 compares key parameter values (CLTP and CPPU) affecting FAC. Based on experience at pre CPPU operating conditions and previous

FAC modeling results, CPPU operating conditions will result in the need for additional FAC inspections.

The increased MS and FW flow rates at CPPU conditions do not significantly affect the potential for FAC in these systems. Increases in the low measured wear rates are expected to increase proportionately with flow. Operation under CPPU conditions will require additional focus for the FAC inspection program for the Main Steam Drains, Moisture Separator Drains, and the Turbine Cross Around System piping. The Extraction Steam System piping at VYNPS is constructed of FAC resistant material.

The reactor internals inspection and FAC programs do not significantly change for CPPU. In addition, the Maintenance Rule provides oversight for the other mechanical and electrical components, important to plant safety, to guard against age-related degradation.

10.8 NRC AND INDUSTRY COMMUNICATIONS

NRC and industry communications could affect the plant design and safety analyses. However, as stated in Section 6.8, all of the systems significantly affected by CPPU are addressed in this report. In addition, all of the plant safety analyses affected by CPPU are addressed in this report. As a result, evaluations of plant design and safety analyses affected by the communications in place are inherently included in the plant-specific CPPU assessments. Therefore, it is not necessary to review prior dispositions of NRC and industry communications and no additional information is required in this area.

10.9 EMERGENCY AND ABNORMAL OPERATING PROCEDURES

Emergency and abnormal operating procedures can be affected by CPPU. Some of the EOPs variables and limit curves depend upon the value of rated reactor power. Some Abnormal Operating Procedures (AOPs) may be affected by plant modifications to support the higher power level. (Note: The AOPs at VYNPS consist of the Operational Transient (OT) and Off Normal (ON) series procedures.) The topics addressed in this section are:

Topic	CPPU Disposition	VYNPS Result
Emergency operating procedures	[[	
Abnormal operating procedures		]]

EOPs include variables and limit curves, defining conditions where operator actions are indicated. Some of these variables and limit curves depend upon the RTP value. Changing some of the variables and limit curves requires modifying the values in the EOPs and updating the VYNPS support documentation. EOP curves and limits may also be included in the SPDS and will be updated accordingly.

The charts and tables used by the operators to perform the EOPs are reviewed for any required changes prior to each core reload. The EOPs will be reviewed for any changes required to

implement CPPU. The operators will receive training on these procedures as described in Section 10.6.

AOPs include event based operator actions. Some of these operator actions may be influenced by plant modifications required to support the increase in RTP. Changing some of the operator actions may require modifications to the AOPs and updating the VYNPS support documentation. The plant AOPs will be reviewed for any effects of CPPU. One AOP will change as a result of modifications to equipment, i.e., the required action following an RFP trip will be changed to reflect the installation of automatic recirculation system runback. Some of the setpoints used in the AOPs will change due to the CPPU. The operators will receive training on these procedures as described in Section 10.6.

Table 10-1
VYNPS Liquid Line Breaks

	Change Due to CPPU		
Break Location	**Mass Release**	**Pressure**	**Temperature**
RWCU Breaks in Reactor Building	Higher / Lower [1.]	Slightly Higher [2.]	Higher / Lower [3.]
Combined RWCU / FW Line Breaks in Steam Tunnel	Higher	Slightly Higher [2.]	Higher [3.]
Combined MS / FW Line Breaks in Steam Tunnel	Higher	Higher [2.]	Higher [3.]

Notes:

1. Changes in integrated mass varied between +4.4% and -5.4%, depending on break location.
2. Within design limits.
3. Assessed as part of EQ.

Table 10-2
VYNPS Equipment Qualification for CPPU

Parameter	CPPU Effect
Inside Containment – Normal Radiation Dose	Increase proportional to uprate
Inside Containment – Normal Temperature	< 1°F increase
Inside Containment – Accident Radiation Dose	≤ 17% increase
Inside Containment – Accident Peak Temperature	No peak temperature increase
Inside Containment – Accident Peak Pressure	No peak pressure increase
Outside Containment – Normal Radiation Dose	Increase proportional to uprate
Outside Containment – Normal Temperature	< 1°F increase
Outside Containment – Accident Radiation Dose	≤ 17% increase
Outside Containment – Accident Temperature	HELB Increase: < 50°F LOCA Increase: ~7°F
Outside Containment – Accident Pressure	No significant change or effect

Table 10-3
Comparison of CLTP CDF vs CPPU CDF

Accident Class	Description	CDF (1/yr)	
		CLTP Value	CPPU Value
ID	Transient sequences with loss of all injection. Core damage occurs with the reactor at low pressure.	3.27E-06	3.27E-06
IA	Transient sequences with loss of all high pressure injection and failure to depressurize. Core damage occurs with the reactor at high pressure.	1.59E-06	1.59E-06
IBL	Late SBO. Core cooling is maintained by HPCI/RCIC until batteries deplete.	7.87E-07	7.90E-07
IIA	Transient sequence with loss of all containment heat removal. Core damage is caused by containment failure.	5.12E-07	5.13E-07
IVL	ATWS sequences where core damage occurs due to overpressure failure of the Reactor Coolant System.	3.08E-07	3.09E-07
IVA	ATWS sequences where core damage is caused by containment failure.	2.82E-07	3.93E-07
IBE	Early SBO sequences. Core damage occurs due to early failure of HPCI and RCIC.	2.65E-07	2.66E-07
IIV	Transient sequences where the main condenser and RHR fail, and the torus vent opens for containment pressure relief. Core damage occurs when ECCS systems fail NPSH, due to failure to reclose the vent.	2.43E-07	2.43E-07
IIIC	LOCA sequences with loss of injection. Core damage occurs with the reactor at low pressure.	2.14E-07	2.17E-07
IED	Early SBO sequences caused by failure of DC-1 and DC-2.	9.85E-08	9.91E-08
V	Containment Bypass sequences. (Interfacing systems LOCA and LOCA outside of containment.)	5.32E-08	5.32E-08
IIIB	SLOCA or MLOCA sequences for which the reactor cannot be depressurized prior to core damage occurring.	5.05E-08	2.43E-07
IIL	Loss of containment heat removal with RPV breach but no initial core damage; core damage after containment failure.	4.76E-08	4.76E-08

Table 10-4
Results of VYNPS CPPU PSA Sensitivity Cases

Parameter ID	CLTP PSA	CPPU	Case #1	Case #2	Case #3	Case #4	Case #5	Case #6	Case #7
Turbine Trip Initiating Event (IE)	5.50E-01	5.70E-01	6.50E-01	CLTP PSA VALUE	CLTP PSA VALUE	1.10E+00	1.10E+00	1.10E+00	5.70E-01
MSIV Closure IE	1.80E-01	CLTP PSA VALUE	CLTP PSA VALUE	2.80E-01	2.80E-01	CLTP PSA VALUE	CLTP PSA VALUE	CLTP PSA VALUE	CLTP PSA VALUE
LLOCA IE	2.40E-05	CLTP PSA VALUE	CLTP PSA VALUE	CLTP PSA VALUE	CLTP PSA VALUE	CLTP PSA VALUE	CLTP PSA VALUE	CLTP PSA VALUE	4.80E-05
SCBASE (SORV)	1.08E-02	1.24E-02	1.30E-02	1.24E-02	1.30E-02	1.24E-02	1.30E-02	2.16E-02	1.24E-02
HEPs	various	MAAP-based(1)	Timings x 0.8(2)	MAAP-based(1)	Timings x 0.8(2)	MAAP-based(1)	MAAP-based(1)	MAAP-based(1)	MAAP-based(1)
CDF:	7.77E-06	8.10E-06	8.2242E-06	8.3532E-06	8.4728E-06	8.3288E-06	8.3344E-06	8.4149E-06	8.1184E-06
delta CDF:	-	3.3E-07	4.54E-07	5.83E-07	7.03E-07	5.59E-07	5.64E-07	6.45E-07	3.48E-07

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(1) "MAAP-based" indicates that the HEPs were re-calculated based on operator allowable timing adjustments determined from MAAP runs performed for the VYNPS CPPU.

(2) "Timings x 0.8" indicates that the HEPs were re-calculated based on operator allowable adjustments that assumed a 20% drop (reflection of the VYNPS CPPU power increase) in the CLTP timings for all actions.

Table 10-4
Results of VYNPS CPPU PSA Sensitivity Cases

Parameter ID	CLTP PSA	CPPU	Case #1	Case #2	Case #3	Case #4	Case #5	Case #6	Case #7
LERF:	2.23E-06	2.34E-06	2.3503E-06	2.3998E-06	2.3981E-06	2.4170E-06	2.4185E-06	2.4397E-06	2.3568E-06
delta LERF:	-	1.1E-07	1.20E-07	1.70E-07	1.68E-07	1.87E-07	1.88E-07	2.09E-07	1.27E-07

Notes:

Sensitivity #1: This sensitivity case modifies the estimates calculated for the CPPU for the Turbine Trip frequency, the SORV probability, and HEPs.

The base CPPU quantification calculated a revised Turbine Trip frequency using an approach that considered the various contributions to the Turbine Trip frequency. This sensitivity case calculates the Turbine Trip frequency using a different approach. This approach assumes an additional turbine trip is experienced in the first year following start-up in the CPPU condition. The change in the long-term average of the Turbine Trip frequency is calculated as follows for this sensitivity case:

- Base long-term Turbine Trip frequency is 0.55/yr
- 10 years is used as the "long-term" data period
- End of 10 years does not reach the end-of-life portion of the bathtub curve
- Model beginning-of-life portion of bathtub curve by assuming one additional turbine trip the first year after start-up in the CPPU condition
- Revised Turbine Trip frequency for this sensitivity case is calculated as:

  $T_{NEW} = ((10 \times 0.55) + 1) / 10 = 0.65/yr$

The base CPPU quantification calculated a revised SORV probability based on reviewing MAAP runs and counting SRV cycles for the CLTP and CPPU conditions. The CLTP base SORV probability is re-calculated in this sensitivity case by simply applying a 1.2 factor reflective of 20% CPPU:

$$SCBASE_{NEW} = 1.08E\text{-}2 \times 1.2 = 1.30E\text{-}2$$

In addition, this sensitivity case revises the HEPs by applying a 20% reduction (to reflect the 20% CPPU) uniformly for all the time allowable estimates.

Sensitivity #2: This sensitivity case conservatively assumes that the potential effect on transient initiator frequencies is manifested in the MSIV Closure initiator frequency and not the Turbine Trip frequency. The CLTP MSIV Closure frequency of 1.80E-2 is revised in this sensitivity case in the same manner as that discussed in Sensitivity Case #1 (i.e., an additional trip is assumed in the first year after start-up at CPPU RTP):

$$TMS_{NEW} = ((10 \times 0.18) + 1)/10 = 0.28/yr$$

All other parameters are maintained the same as the CPPU base case.

Sensitivity #3: This sensitivity case combines the SCBASE and HEP changes of Sensitivity Case #1 with the Transient With MSIV Closure (TMS) change of Sensitivity Case #2.

Sensitivity #4: This sensitivity case conservatively assumes that the potential effect on the transient initiator frequencies is manifested as a doubling of the Turbine Trip initiator frequency. All other parameters are maintained the same as the CPPU base case.

Sensitivity #5: This sensitivity case combines the changes of Sensitivity Case #4 with a 1.2x increase in the base SORV probability. All other parameters are maintained the same as the CPPU base case.

Sensitivity #6: This sensitivity case conservatively doubles both the Turbine Trip initiator frequency and the stuck-open relief valve probability. All other parameters are maintained the same as the CPPU base case.

Sensitivity #7: The CPPU base quantification does not modify the DBA LOCA frequency. Acknowledging that the increased flow rates at CPPU can result in increased piping erosion/corrosion rates, this sensitivity case conservatively doubles the LLOCA initiator frequency. All other parameters are maintained the same as the CPPU base case

Table 10-5
Assessment Of Key Operator Actions

Action ID	Action Description	Allowable Action Time: Current PSA (CLTP)	Allowable Action Time: CPPU	CLTP HEP	CPPU HEP	Comment
AOPHRIFL	OPERATOR FAILS TO MANUALLY INITIATE HPCI AND RCIC SYSTEMS	66 min. (Trans) 35 min. (Medium LOCA (MLOCA))	48.2 min. (Trans) 24.1 min. (MLOCA)	2.1E-3	2.1E-3	Time to 1/3 core height w/o injection (66 min. for transients; ~35 min. for MLOCAs, estimated at half the time of transient case). MAAP cases VYEPU2 and 2x show a ~27% drop in allowable timing. HEP dominated by manipulation error; HEP value is the same for either case.
EOPADMFL	OPERATOR FAILS TO MANUALLY OPEN SRVS FOR MEDIUM LOCA	33 min.	24.1 min.	9.1E-4	4.9E-3	Time to 1/3 core height for a MLOCA w/o injection (estimated at half the time of transient case in VYNPS PSA).
EOPADSFL	OPERATOR FAILS TO MANUALLY OPEN SRVS FOR TRANSIENT/SMALL LOCA	66 min.	48.2 min.	2.1E-4	2.1E-4	Time to 1/3 core height for a Transient w/o injection. MAAP cases VYEPU2 and 2x show a ~27% drop in allowable timing.
EOPED1FL	OPERATOR FAILS TO MANUALLY OPEN SRVS (ATWS, HCTL EXCEEDED)	16 min.	14.4 min.	2.4E-3	4.2E-3	Time to reach HCTL RPV ED curve for an isolation ATWS scenario. MAAP case VYEPU6h shows CPPU timing to be 14.4 min.

Table 10-5
Assessment Of Key Operator Actions

Action ID	Action Description	Allowable Action Time		CLTP HEP	CPPU HEP	Comment
		Current PSA (CLTP)	CPPU			
EOPMD1FL	OPERATOR FAILS TO MANUALLY INITIATE DEPRESSURIZATION FOR VAPOR SUPPRESSION DURING MLOCA	10 min.	10 min.	9.9E-2	9.9E-2	Time to containment failure for MLOCA scenario with stuck open WW-DW vacuum breakers; 10 min. time based on NUREG/CR-4594. Timing is conservative. MAAP cases VYEPU8b and 8bx show that containment failure does not occur with a stuck open vacuum breaker. The 10 min. time frame is maintained.
EOPSM1FL	OPERATOR FAILS TO DEPRESSURIZE FOR VAPOR SUPPRESSION DURING a SMALL LOCA (SLOCA)	21 min.	21 min.	4.6E-3	4.6E-3	Time to PSA containment failure pressure for SLOCA scenario with stuck open WW-DW vacuum breakers. Timing is conservative. MAAP cases VYEPU8a and 8ax show that containment failure does not occur with a stuck open vacuum breaker. The 21 min. time frame is maintained.
HOPALTINJFL	OPERATOR FAILS TO ALIGN ALTERNATE INJECTION USING CS OR CONDENSATE TRANSFER WITH SUCTION FROM CST	9-10 hrs.	7.6 hrs.	3.1E-2	3.1E-2	Time to reach 200F temperature in SP for transients or SLOCAs with no containment heat removal. This is the time at which CS suction from the SFP will need to be replaced by another source. MAAP cases VYEPU9 and 9x show a ~20% drop in allowable timing.
HOPCRPFL	OPERATOR FAILS TO START A CRD PUMP	2 hrs. (after many hrs. into the event)	96 min. (after many hrs. into the event)	2.6E-4	2.6E-4	Time to 1/3 core height following loss of all injection at containment failure during a loss of containment heat removal accident. This time frame begins many hours after plant trip. The timing is adjusted downward by 20%.

Table 10-5
Assessment Of Key Operator Actions

Action ID	Action Description	Allowable Action Time Current PSA (CLTP)	Allowable Action Time CPPU	CLTP HEP	CPPU HEP	Comment
IA01FL	SIMPLE ACTION (OPEN DOOR) IN 10 MINUTES FOR FLOOD EVENT MITIGATION.	10 min.	10 min.	1.0E-1	1.0E-1	Timing based on internal flooding issues and not directly on reactor power.
IA12FL	SIMPLE ACTION (OPEN DOOR) IN 10 TO 20 MINUTES FOR FLOOD EVENT MITIGATION.	20 min.	20 min.	1.0E-2	1.0E-2	Timing based on internal flooding issues and not directly on reactor power.
IA23FL	SIMPLE ACTION (OPEN DOOR/CLOSE VALVE) IN 20 TO 30 MINUTES FOR FLOOD EVENT MITIGATION.	30 min.	30 min.	1.0E-3	1.0E-3	Timing based on internal flooding issues and not directly on reactor power.
IA4PFL	SIMPLE ACTION (OPEN DOOR/CLOSE VALVE/STOP PUMP) AFTER 30 MINUTES FOR FLOOD EVENT MITIGATION, LOWER BOUND HEP.	>30 min.	>30 min.	1.0E-4	1.0E-4	Timing based on internal flooding issues and not directly on reactor power.
IABASE	OPERATOR INHIBITS ADS (ATWS)	6.2 min.	5.4 min.	1.6E-3	3.3E-3	Time to reach RPV level low-low set point for ATWS without injection plus 2 minutes (for ADS timer). MAAP cases VYEPU6a and 6ax show a ~15% drop in the time to Level 2.

Table 10-5
Assessment Of Key Operator Actions

Action ID	Action Description	Allowable Action Time		CLTP HEP	CPPU HEP	Comment
		Current PSA (CLTP)	CPPU			
IOPSLMCF	OPERATOR FAILS TO INITIATE SLC SYSTEM GIVEN MAIN CONDENSER FAILED	6 min.	5.3 min.	5.7E-2	8.1E-2	Time for SP temperature to reach 110F (EOP BIIT) during an ATWS scenario with condenser unavailable and SP temperature initially at 70F. MAAP cases VYEPU6a and 6ax show a ~12% drop in allowable timing.
IOPSLMCS	OPERATOR FAILS TO INITIATE SLC SYSTEM GIVEN MAIN CONDENSER SUCCESS	60 min.	60 min.	1.2E-3	1.2E-3	Time for SP temperature to reach 110F (EOP BIIT) during an ATWS scenario with condenser available and SP temperature initially at 70F. Timing is conservative. MAAP cases VYEPU6i and 6ix show that 110°F in the pool is not reached for many hours. The 60 min. timing is maintained.
ISOPLLFL	OPERATOR FAILS TO ISOLATE PATH DURING LARGE LOCA (LLOCA)	20 min.	19 min.	3.1E-1	3.4E-1	Time to 1/3 core height for a LLOCA w/o injection. MAAP cases VYEPU7c and 7cx show a ~5% drop in allowable timing.
JOPFIS01	OPERATOR FAILS TO INITIATE FIRE SYSTEM AND J.D. DIESEL FOR AI	At least 1 hr.	Approx. 1 hr.	1.0E-1	1.0E-1	Time to reach 1/3 core height after loss of injection that had been initially running for 1-4 hours. HEP is a screening value that is not affected by CPPU.
KOPACTFL	OPERATOR FAILS TO INITIATE SUPPRESSION POOL COOLING	>24 hrs	>24 hrs	1.0E-6	1.0E-6	Time to reach PSA containment ultimate pressure for a transient loss of containment heat removal accident. Very long time frame available. HEP based on industry accepted value for this action. This HEP is not affected by CPPU.

Table 10-5
Assessment Of Key Operator Actions

Action ID	Action Description	Allowable Action Time		CLTP HEP	CPPU HEP	Comment
		Current PSA (CLTP)	CPPU			
KOPATWS1FL	OPERATOR INITIATES RHR IN SUPPRESSION POOL COOLING (SPC) MODE (ATWS)	15 min.	12 min.	6.2E-3	6.9E-3	Conservative time estimate, loosely based on the NPSH problems in the pool and the fact that the operators will be performing many actions in a short time frame. Allowable timing reduced by 20%.
LCATWS1FL	OPERATOR TERMINATES AND PREVENTS ALL INJECTION SLC, CRD, AND RCIC BEFORE RPV DEPRESSURIZATION (ATWS)	15 min.	14.4 min.	1.3E-2	1.5E-2	Time to reach HCTL RPV ED curve for an isolation ATWS scenario. MAAP case VYEPU6h shows CPPU timing to be 14.4 min.
LCATWS2FL	OPERATOR LOWERS RPV WATER LEVEL TO TAF FOR POWER CONTROL AND RESTORES RPV LEVEL AFTER SLC INJECTION	17 min.	13.6 min.	6.1E-3	1.6E-2	Time at which if level is not lowered to TAF that pool temperature will reach 240F (isolation ATWS scenario). A drop of 20% available timing is assumed.
LIATWS1FL	OPERATOR RESTORES Low Pressure Injection (LPI) POST RPV DEPRESSURIZATION (ATWS)	15 min.	12 min.	1.4E-2	2.1E-2	Time to boil off level from TAF to 1/3 core height during an isolation ATWS scenario with level control down to TAF and Emergency Depressurization, plus a couple minutes for regaining water level. MAAP cases VYEPU6c and 6cx show a ~22% drop in time from TAF to 1/3 core height.

Table 10-5
Assessment Of Key Operator Actions

Action ID	Action Description	Allowable Action Time: Current PSA (CLTP)	Allowable Action Time: CPPU	CLTP HEP	CPPU HEP	Comment
MOPTVFL1	OPERATOR FAILS TO RECOGNIZE THE NEED TO VENT TORUS FOR PRESSURE REDUCTION	~5 hrs.	~5 hrs.	1.3E-3	1.3E-3	Time from HI DW signal to 65 psi in containment during a loss of containment heat removal scenario. Timing is conservative. MAAP cases VYEPU9a and 9ax show time available is approximately 22 hrs and drops a few hours for the CPPU condition. The 5 hr time allowable is maintained.
OPMSIVBP	OPERATOR BYPASSES MSIV ISOLATION INTERLOCKS (ATWS)	4 min.	3.4 min.	3.1E-2	4.9E-2	Time to reach RPV level low-low set point for ATWS w/o injection. MAAP cases VYEPU6a and 6ax show a ~15% drop in the time to RPV level L2.
QOP001FL	OPERATOR FAILS TO INITIATE/CONTROL FEEDWATER/ CONDENSATE	28 min.	21 min.	3.1E-3	3.1E-3	Conservatively taken as the time to TAF + a few additional minutes for depressurization time. MAAP cases VYEPU1c and 1cx show a ~29% drop in the time to TAF.
QOP003FL	OPERATOR FAILS TO OPEN MOV 64-31	30 min.	30 min.	2.0E-3	2.0E-3	Time to deplete hotwell for transient with MSIV closure, using Condensate for RPV injection, and hotwell makeup from CST is 600 gpm, and hotwell inventory initially at 30,000 gallons. The 30 min. time frame is maintained.

Table 10-5
Assessment Of Key Operator Actions

Action ID	Action Description	Allowable Action Time: Current PSA (CLTP)	Allowable Action Time: CPPU	CLTP HEP	CPPU HEP	Comment
RMOPATWS	OPERATOR REOPENS MSIVs AND RESTORES CONDENSER FOR CONTAINMENT HEAT REMOVAL (ATWS)	25 min.	20 min.	2.1E-1	7.3E-1	Time allowable based on time for pool temperature to reach 240°F. A drop of 20% available timing is assumed.
TOPSSW02	OPERATOR FAILS TO INITIATE REQUIRED SW PUMPS	2 hrs.	2 hrs.	2.0E-3	2.0E-3	Timing based on judgment and component heat-up (one SW pump already running). Timing not directly related to reactor power.
UA23FL	SIMPLE ACTION (OPEN DOOR) IN 20 TO 30 MINUTES FOR FLOOD EVENT MITIGATION (INDEPENDENT OF Initial Operator Action (IOA)).	30 min.	30 min.	1.0E-3	1.0E-3	Timing based on internal flooding issues and not directly on reactor power.
UA3PFL	SIMPLE ACTION (OPEN DOOR/CLOSE VALVE) AFTER 30 MINUTES FOR FLOOD EVENT MITIGATION (INDEPENDENT OF IOA).	>30 min.	>30 min.	5.0E-4	5.0E-4	Timing based on internal flooding issues and not directly on reactor power.
UAHDFL	ULTIMATE FLOODING ACTION WITH HIGH DEPENDENCE (HD) ON THE INITIAL ACTION.	>10 to <20 min.	>10 to <20 min.	1.5E-1	1.5E-1	Timing based on internal flooding issues and not directly on reactor power.

Table 10-5
Assessment Of Key Operator Actions

Action ID	Action Description	Allowable Action Time: Current PSA (CLTP)	Allowable Action Time: CPPU	CLTP HEP	CPPU HEP	Comment
UALDFL	ULTIMATE FLOODING ACTION WITH MODERATE DEPENDENCE (MD) ON THE INITIAL ACTION.	>20 to <40 min.	>20 to <40 min.	5.0E-2	5.0E-2	Timing based on internal flooding issues and not directly on reactor power.
UAMDFL	ULTIMATE FLOODING ACTION WITH LOW DEPENDENCE (LD) ON THE INITIAL ACTION.	>40 min.	>40 min.	1.5E-1	1.5E-1	Timing based on internal flooding issues and not directly on reactor power.
UOPACM1FL	OPERATOR FAILS TO INITIATE ALTERNATE COOLING	12 hrs.	9.6 hrs.	3.0E-2	3.0E-2	The 12 hr. available time frame used in the base PSA is based on judgment. A drop of 20% in available timing is assumed; however, the HEP remains unchanged.
VDOPERROR2	OPERATOR FAILS TO DEPRESSURIZE DURING ADDITIONAL ONE HOUR	1 hr.	1 hr.	5.0E-1	5.0E-1	Time from core damage initiation until the time at which RPV flooding will not be able to prevent RPV breach. This Level 2 PSA timing issue is based on industry studies and judgment and is not directly linked to reactor power. The 1 hr. time frame is maintained.
VOPRBC01	OPERATOR FAILS TO START RBCCW PUMP	30 min.	30 min.	3.3E-2	3.3E-2	Time for LPCI pump heat up and seal failure. Timing not directly related to reactor power.

Table 10-5
Assessment Of Key Operator Actions

Action ID	Action Description	Allowable Action Time: Current PSA (CLTP)	Allowable Action Time: CPPU	CLTP HEP	CPPU HEP	Comment
VROPERROR3	OPERATOR FAILS TO ALIGN RHRSW INJECTION TO RPV	15 min.	11.6 min.	2.2E-1	3.6E-1	Time to 1/3 core height during an isolation ATWS scenario. This conservative timing is used for all applications of RHRSW crosstie. MAAP cases VYEPU6a and 6ax show a ~23% drop in allowable timing.
WOPTBC01	OPERATOR FAILS TO START TBCCW PUMP	30 min.	30 min.	3.7E-3	3.7E-3	Time for BOP component load heat up. Timing not directly related to reactor power.
XOPRSAFL	OPERATOR FAILS TO RESET C.1.1A & C.1.1B FOLLOWING LOSS OF POWER	10 min.	10 min.	8.0E-2	8.0E-2	Conservative estimate of the time for bleed down of the instrument air receivers. Timing not directly related to reactor power.
YOPAC1FL	OPERATOR FAILS TO CLOSE VERNON TIE BREAKERS	28 min.	21 min.	1.2E-3	1.6E-3	Conservatively taken as the time to TAF + a few additional minutes for depressurization time. MAAP cases VYEPU1c and 1cx show a ~29% drop in the time to TAF.
YOPVACFL	OPERATOR FAILS TO RESTORE MCC-8B TO THE MG SET AFTER LNP	30 min.	30 min.	1.0E-1	1.0E-1	Conservative time estimate based loosely on the time to 1/3 core height during a loss of injection scenario, and recognizing that the operators will be focusing on other makeup restoration activities. HEP is a screening value that is not affected by CPPU.

Table 10-6
Estimate of the Effect on Fire CDF due to CPPU

Dominant Fire Scenarios:

Fire Scenario	CDF	Breakdown Based on Review of VY IPEEE RISKMAN Results					
		Loss of Coolant Makeup Accident Sequences		Loss of Decay Heat Removal Accident Sequences		ATWS Accident Sequences	
		%	CDF	%	CDF	%	CDF
FCVS2F	1.30E-05	95	1.24E-05	5	6.50E-07	negligible	0.00E+00
FCR	5.70E-06	95	5.42E-06	5	2.85E-07	negligible	0.00E+00
FSG24	4.40E-06	99	4.36E-06	1	4.40E-08	negligible	0.00E+00
FSGW1	3.70E-06	92	3.40E-06	8	2.96E-07	negligible	0.00E+00
FCVBAT	3.20E-06	95	3.04E-06	5	1.60E-07	negligible	0.00E+00
FSGW3	3.20E-06	87	2.78E-06	13	4.16E-07	negligible	0.00E+00
FRB4CR	2.20E-06	98	2.16E-06	2	4.40E-08	negligible	0.00E+00
FRB3MC	2.10E-06	99	2.08E-06	1	2.10E-08	negligible	0.00E+00
FSGWT	2.10E-06	87	1.83E-06	13	2.73E-07	negligible	0.00E+00
FRB3CL	1.90E-06	98	1.86E-06	2	3.80E-08	negligible	0.00E+00
FSGET	1.80E-06	99	1.78E-06	1	1.80E-08	negligible	0.00E+00
FCVCL	1.40E-06	95	1.33E-06	5	7.00E-08	negligible	0.00E+00
FRB3TR	1.10E-06	100	1.10E-06	0	0.00E+00	negligible	0.00E+00
	4.58E-05		4.35E-05		2.32E-06		0.00E+00
			(95%)		(5%)		(0%)
	This is taken as an approximation of the VY fire CDF for the purpose of the EPU		This is the portion of the VY fire risk due to loss of coolant makeup core damage accidents		This is the portion of the VY fire risk due to loss of containment heat removal core damage accidents.		This is the portion of the VY fire risk due fire risk due to ATWS core damage accidents

Estimation of Change in VY Fire Risk Due to EPU:

Delta Fire CDF (%) = [(95% x 4.58E-5 x (1 + 0.015)) + (5% x 4.58E-5 x (1 + 0.00)) + (0% x 4.58E-5 x (1 + 0.10)) - 4.58E-5] / 4.58E-5

$\boxed{1.4\%}$

Where: - The 95% term represents the fraction of the internal fires CDF due to early loss of coolant makeup accidents.

- The 0.015 term represents the approximate 3% increase manifested in the internal events for loss of coolant makeup accidents multiplied by 0.5% (to account for the fact that the fire scenarios are dominated by fire-induced failures and not so much impacted by operator actions timings as the internals events).
- The 5% term represents the fraction of the internal fires CDF due to loss of decay heat removal accidents.
- The 0.00 term represents the approximate 0% increase manifested in the internal events for loss of decay heat removal accidents.
- The 0% term represents the fraction of the internal fires CDF due to ATWS accidents.
- The 0.10 term represents the approximate 20% increase manifested in the internal events for ATWS accidents multiplied by 0.5% (to account for the fact that the fire scenarios are dominated by fire-induced failures and not so much impacted by operator actions timings as the internals events).

Table 10-7
VYNPS FAC Parameter Comparison for CPPU

Parameter	CLTP RTP Range of Values	CPPU RTP Range of Values
Steam Flow (lbm/hr)	6,458,000	7,906,000
Main Steam Quality (%)	99.9	99.9
Main Steam Velocity (ft/sec)	146	181
Feedwater Piping Operating Temperatures (°F)	298 to 375	312 to 392
Feedwater Flow (lbm/hr)	6,430,000 / 6,498,000	7,878,000 / 7,946,000
Feedwater Velocity (ft/sec)	12.6 to 18.2	15.5 to 22.4

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Docket No. 50-271
BVY 03-80

Attachment 7

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 263

Extended Power Uprate

Justification for Exception to Large Transient Testing

JUSTIFICATION FOR EXCEPTION TO LARGE TRANSIENT TESTING

Background

The basis for the Constant Pressure Power Uprate (CPPU) request was prepared following the guidelines contained in the NRC approved, General Electric (GE) Company Licensing Topical Report for Constant Pressure Power Uprate (CLTR) Safety Analysis: NEDC-33004P-A Rev. 4, July 2003. The NRC staff did not accept GEs proposal for the generic elimination of large transient testing (i.e., Main Steam Isolation Valve (MSIV) closure and turbine generator load rejection) presented in NEDC-33004P Rev. 3. Therefore, on a plant specific basis, Vermont Yankee Nuclear Power Station (VYNPS) is taking exception to the large transient tests; MSIV closure and turbine generator load rejection.

The CPPU methodology, maintaining a constant pressure, simplifies the analyses and plant changes required to achieve uprated conditions. Although no plants have implemented an Extended Power Uprate (EPU) using the CLTR, thirteen plants have implemented EPUs without increasing reactor pressure.

- Hatch Units 1 and 2 (105% to 113% of Original Licensed Thermal Power (OLTP))
- Monticello (106% OLTP)
- Muehleberg (i.e., KKM) (105% to 116% OLTP)
- Leibstadt (i.e., KKL) (105% to 117% OLTP)
- Duane Arnold (105% to 120% OLTP)
- Brunswick Units 1 and 2 (105% to 120% OLTP)
- Quad Cities Units 1 and 2 (100% to 117% OLTP)
- Dresden Units 2 and 3 (100% to 117% OLTP)
- Clinton (100% to 120%)

Data collected from testing responses to unplanned transients for Hatch Units 1 and 2 and KKL plants has shown that plant response has consistently been within expected parameters.

Entergy believes that additional MSIV closure and generator load rejection tests are not necessary. If performed, these tests would not confirm any new or significant aspect of performance that is not routinely demonstrated by component level testing. This is further supported by industry experience which has demonstrated plant performance, as predicted, under EPU conditions. VYNPS has experienced generator load rejections from 100% current licensed thermal power (see VYNPS Licensee Event Reports (LER) 91-005, 91-009, and 91-014). No significant anomalies were seen in the plant's response to these events. Further testing is not necessary to demonstrate safe operation of the plant at CPPU conditions. A Scram from high power level results in an unnecessary and undesirable transient cycle on the primary system. In addition, the risk posed by intentionally initiating a MSIV closure transient or a generator load rejection, although small, should not be incurred unnecessarily.

VYNPS Response to Unplanned Transients:

VYNPS experienced an unplanned Generator Load Rejection from 100% power on 04/23/91. The event included a loss of off site power. A reactor scram occurred as a result of a Generator/Turbine trip on generator load reject due to the receipt of a 345 KV breaker failure

signal. This was reported to the NRC in LER 91-009, dated 05/23/91. No significant anomalies were seen in the plant's response to this event. VYNPS also experienced the following unplanned generator load rejection events:

- On 3/13/91 with reactor power at 100% a reactor scram occurred as a result of turbine trip on generator load reject due to a 345KV Switchyard Tie Line Differential Fault. This event was reported to the NRC in LER 91-005, dated 4/12/91.
- On 6/15/91 during normal operation with reactor power at 100% a reactor scram occurred due to a Turbine Control Valve Fast Closure on Generator Load Reject resulting from a loss of the 345KV North Switchyard bus. This event was reported to the NRC in LER 91-014, dated 7/15/91.

No significant anomalies were seen in the plant's response to these events. Transient experience at high powers and for a wide range of power levels at operating BWR plants has shown a close correlation of the plant transient data to the predicated response.

Based on the similarity of plants, past transient testing, past analyses, and the evaluation of test results, the effects of the CPPU RTP level can be analytically determined on a plant specific basis. The transient analysis performed for the VYNPS CPPU demonstrates that all safety criteria are met and that this uprate does not cause any previous non-limiting events to become limiting. No safety related systems were significantly modified for the CPPU, however some instrument setpoints were changed. The instrument setpoints that were changed do not contribute to the response to large transient events. No physical modification or setpoint changes were made to the SRVs. No new systems or features were installed for mitigation of rapid pressurization anticipated operational occurrences for this CPPU. A Scram from high power level results in an unnecessary and undesirable transient cycle on the primary system. Therefore, additional transient testing involving scram from high power levels is not justifiable. Should any future large transients occur, VYNPS procedures require verification that the actual plant response is in accordance with the predicted response. Existing plant event data recorders are capable of acquiring the necessary data to confirm the actual versus expected response.

Further, the important nuclear characteristics required for transient analysis are confirmed by the steady state physics testing. Transient mitigation capability is demonstrated by other equipment surveillance tests required by the Technical Specifications. In addition, the limiting transient analyses are included as part of the reload licensing analysis.

MSIV Closure Event

Closure of all MSIVs is an Abnormal Operational Transient as described in Chapter 14 of the VYNPS Updated Final Safety Analysis Report (UFSAR). The transient produced by the fast closure (3.0 seconds) of all main steam line isolation valves represents the most severe abnormal operational transient resulting in a nuclear system pressure rise when direct scrams are ignored. The Code overpressure protection analysis assumes the failure of the direct isolation valve position scram. The MSIV closure transient, assuming the backup flux scram verses the valve position scram, is more significant. This case has been re-evaluated for CPPU with acceptable results.

The CLTR states that: "The same performance criteria will be used as in the original power ascension tests, unless they have been replaced by updated criteria since the initial test program." The original MSIV closure test allowed the scram to be initiated by the MSIV position switches.

As such, if the original MSIV closure test were re-performed, the results would be much less significant than the MSIV closure analysis performed by GE for CPPU.

The original MSIV closure test was intended to demonstrate the following:

1. *Determine reactor transient behavior during and following simultaneous full closure of all MSIVs.*

   Criteria:
   a) *Reactor pressure shall be maintained below 1230 psig.*
   b) *Maximum reactor pressure should be 35 psi below the first safety valve setpoint. (This is margin for safety valve weeping).*

2. *Functionally check the MSIVs for proper operation and determine MSIV closure time.*

   Criteria:
   a) *Closure time between 3 and 5 seconds.*

Item 1: Reactor Transient Behavior

For this event, the closure of the MSIVs cause a vessel pressure increase and an increase in reactivity. The negative reactivity of the scram from MSIV position switches should offset the positive reactivity of the pressure increase such that there is a minimal increase in heat flux. Therefore, the thermal performance during the proposed MSIV closure test is much less limiting than any of the transients routinely re-evaluated. CPPU will have minimal impact on the components important to achieving the desired thermal performance. Reactor Protection system (RPS) logic is unaffected and with no steam dome pressure increase, overall control rod insertion times will not be significantly affected. MSIV closure speed is controlled by adjustments to the actuator and is considered very reliable as indicated below.

Reactor Pressure

Due to the minimal nature of the flux transient, the expected reactor pressure rise, Item 1 above, is largely dependent on SRV setpoint performance. At VYNPS all four SRVs are replaced with re-furbished and pre-tested valves each outage. After the outage, the removed valves are sent out for testing and recalibration for installation in the following outage. Over the past ten years there have been twenty five SRV tests performed. In those twenty five tests only one test found the as-found setting outside the Technical Specification (TS) current allowable tolerance of ±3%. This valve was found to deviate by 3.4% of its nominal lift setpoint. Note that this is bounded by the VYNPS design analysis for peak vessel pressure which assumes one of the four SRVs does not open at all (one SRV out of service). Given the historical performance of the VYNPS SRVs along with the design margins, performance of an actual MSIV closure test would provide little benefit for demonstrating vessel overpressure protection that is not already accomplished by the component level testing that is routinely performed, in accordance with the VYNPS TSs.

Because rated vessel steam dome pressure is not being increased and SRV setpoints are not being changed, there is no increase in the probability of leakage after a SRV lift. Since SRV leakage performance is considered acceptable at the current conditions, which match CPPU conditions with respect to steam dome pressure and SRV setpoints, SRV leakage performance should continue to be acceptable at CPPU conditions. An MSIV closure test would provide no

significant additional confirmation of Item 1 performance criteria than the routine component testing performed every cycle, in accordance with the VYNPS TSs.

Item 2: MSIV Closure Time

Since steam flow assists MSIV closure, the focus of Item 2 was to verify that the steam flow from the reactor was not shut off faster than assumed (i.e., 3 seconds). During maintenance and surveillance, MSIV actuators are evaluated and adjusted as necessary to control closure speed, and VYNPS test performance has been good. To account for minor variations in stroke times, the calibration test procedure for MSIV closure (OP 5303) requires an as left fast closure time of 4.0 ±0.2 seconds. The MSIVs were evaluated for CPPU. The evaluation included MSIV closure time and determined that the MSIVs are acceptable for CPPU operation. Industry experience, including VYNPS, has shown that there are no significant generic problems with actuator design. Confidence is very high that steam line closure would not be less than assumed by the analysis.

Other Plant Systems and Components Response

The MSIV limit switches that provide the scram signal are highly reliable devices that are suitable for all aspects of this application including environmental requirements. There is no direct effect by any CPPU changes on these switches. There may be an indirect impact caused by slightly higher ambient temperatures, but the increased temperatures will still be below the qualification temperature. These switches are expected to be equally reliable before and after CPPU.

The Reactor Protection System (RPS) and Control Rod Drive (CRD) components that convert the scram signals into CRD motion are not directly affected by any CPPU changes. Minor changes in pressure drops across vessel components may result in very slight changes in control blade insertion rates. These changes have been evaluated and determined to be insignificant. The ability to meet the scram performance requirement is not affected by CPPU. Technical Specification (TS) requirements for these components will continue to be met.

CPPU Modifications

Feedwater System operation will require operation of all three feed pumps at CPPU conditions (unlike CLTP conditions). Operation of the additional Reactor Feed Pump (RFP) will not affect plant response to an MSIV closure transient. All feedwater pumps receive a trip signal prior to level reaching 177 inches. Overfill of the vessel after a trip would only occur if level exceeded approximately 235.5 inches. Since the feedwater pumps, the High Pressure Coolant Injection (HPCI) turbine, and the RCIC turbine all receive trip signals prior to level reaching 177 inches, a substantial margin exists. VYNPS operating history has demonstrated that this margin greatly exceeds vessel level overshoot during transient events. Based on this, there is adequate confidence that the vessel level will remain well below the main steam lines under CPPU conditions. The HPCI and RCIC pump trip functions are routinely verified as required by TSs and are considered very reliable.

The modification adding a recirculation pump runback following a RFP trip will not affect the plant response to this transient. The reactor scram signal from the MSIV limit switches will result in control rod insertion prior to any manual or automatic operation of the RFPs. Since

control rods will already be inserted, a subsequent runback of the recirculation pumps will not affect the plant response.

The modification (BVY 03-23 "ARTS/MELLLA") to add an additional unpiped Spring Safety Valve (SSV) will not affect the plant response to this transient. The new third SSV will have the same lift setpoint as the two existing SSVs. This transient does not result in an opening of a SSV, nor is credit taken for SSV actuation.

Generator Load Reject Testing

"Generator Load Rejection From High Power Without Bypass" (GLRWB) is an Abnormal Operational Transient as described in Chapter 14 of the VYNPS Updated Final Safety Analysis Report (UFSAR). This transient competes with the turbine trip without bypass as the most limiting overpressurization transient that challenges thermal limits for each cycle. The GLRWB analysis assumes that the transient is initiated by a rapid closure of the turbine control valves. It also assumes that all bypass valves fail to open.

The CLTR states that: "The same performance criteria will be used as in the original power ascension tests, unless they have been replaced by updated criteria since the initial test program." The startup test for generator load reject allowed the select rod insert feature to reduce the reactor power level and, in conjunction with bypass valve opening, control the transient such that the reactor does not scram. Current VYNPS design does not include the select rod insert feature. The plant was also modified to include a scram from the acceleration relay of the turbine control system. Under current plant design, the original generator load reject test can not be re-performed. If a generator load reject with bypass test were performed, the results would be much less significant than the generator load reject without bypass closure analysis performed by GE for CPPU.

The original generator load reject test was intended to demonstrate the following:

1. *Determine and demonstrate reactor response to a generator trip, with particular attention to the rates of changes and peak values of power level, reactor steam pressure and turbine speed.*

   Criteria:
   a. *All test pressure transients must have maximum pressure values below 1230 psig*
   b. *Maximum reactor pressure should be 35 psi below the first safety valve setpoint. (This is margin for safety valve weeping).*
   c. *The select rod insert feature shall operate and in conjunction with proper bypass valve opening, shall control the transient such that the reactor does not scram.*

Due to plant modification discussed above, Criterion c. above would no longer be applicable for a generator load reject test. The generator load reject startup test was performed at 93.7% power; however, a reactor scram occurred during testing and invalidated the test. A design change to initiate an immediate scram on generator load reject was implemented and this startup test was subsequently cancelled since it was no longer applicable.

Item 1 Reactor Response

For a generator load reject with bypass event, given current plant design, the fast closure of the Turbine Control Valves (TCVs) cause a trip of the acceleration relay in the turbine control system. The acceleration relay trip initiates a full reactor scram. The bypass valves open, however, since the capacity of the bypass valves at CPPU is 87%, vessel pressure increases. This results in an increase in reactivity. The negative reactivity of the TCV fast closure scram from the acceleration relay should offset the positive reactivity of the pressure increase such that there is a minimal increase in heat flux. Therefore, the thermal performance during a generator load rejection test would be much less limiting than any of the transients routinely re-evaluated. CPPU will have minimal impact on the components important to achieving the desired thermal performance. Reactor Protection system (RPS) logic is unaffected and with no steam dome pressure increase, overall control rod insertion times will not be significantly affected. A trip channel and alarm functional test of the turbine control valve fast closure scram is performed every three months in accordance with plant technical specifications. This trip function is considered very reliable.

Reactor Pressure

Due to the minimal nature of the flux transient, the expected reactor pressure rise, Criteria a. and b. above, are largely dependent on SRV setpoint performance. Refer to the MSIV closure Reactor Pressure section above for discussion of SRV setpoint performance.

Because rated vessel steam dome pressure is not being increased and SRV setpoints are not being changed, there is no increase in the probability of leakage after a SRV lift. Since SRV leakage performance is considered acceptable at the current conditions, which match CPPU conditions with respect to steam dome pressure and SRV setpoints, SRV leakage performance will continue to be acceptable at CPPU conditions. A generator load rejection test would provide no significant additional confirmation of performance criteria a. and b. than the routine component testing performed every cycle, in accordance with the VYNPS TSs.

Other Plant Systems and Components Response

The turbine control system acceleration relay hydraulic fluid pressure switches that provide the scram signal are highly reliable devices that are suitable for all aspects of this application including environmental requirements. There is no direct effect by any CPPU changes on these pressure switches. These switches are expected to be equally reliable before and after CPPU.

The Reactor Protection System (RPS) and Control Rod Drive (CRD) components that convert the scram signals into CRD motion are not directly affected by any CPPU changes. Minor changes in pressure drops across vessel components may result in very slight changes in control blade insertion rates. These changes have been evaluated and determined to be insignificant. The ability to meet the scram performance requirement is not affected by CPPU. TS requirements for these components will continue to be met.

CPPU Modifications

As previously described, Feedwater System operation will require all three feed pumps at CPPU conditions. Operation of the additional Reactor Feed Pump (RFP) will not affect plant response to this transient. All feedwater pumps receive a trip signal prior to level reaching 177 inches.

Overfill of the vessel after a trip would only occur if level exceeded approximately 235.5 inches. Since the feedwater pumps, the High Pressure Coolant Injection (HPCI) turbine, and the RCIC turbine all receive trip signals prior to level reaching 177 inches, a substantial margin exists. VYNPS operating history has demonstrated that this margin greatly exceeds vessel level overshoot during transient events. Based on this, there is adequate confidence that the vessel level will remain well below the main steam lines under CPPU conditions. The HPCI and RCIC pump trip functions are routinely verified as required by TSs and are considered very reliable.

The modification adding a recirculation pump runback following a RFP trip will not affect the plant response to this transient. The reactor scram signal from turbine control valve fast closure will result in control blade insertion prior to any manual or automatic operation of the RFPs. Since control blades will already be inserted, a subsequent runback of the recirculation pumps will not affect the plant response.

The modification (BVY 03-23) "ARTS/MELLLA") to add an additional unpiped SSV will not affect the plant response to this transient. The new third SSV will have the same lift setpoint of the two existing SSVs. This transient does not result in an opening of a SSV nor is credit taken for SSV actuation.

HP Turbine modification replaces the steam flow path but will not affect the turbine control system hydraulic pressure switches that provide the turbine control valve fast closure scram signal to the RPS system.

Industry Boiling Water Reactor (BWR) Power Uprate Experience

Southern Nuclear Operating Company's (SNOC) application for EPU of Hatch Units 1 and 2 was granted without requirements to perform large transient testing. VYNPS and Hatch are both BWR/4 with Mark 1 containments. Although Hatch was not required to perform large transient testing, Hatch Unit 2 experienced an unplanned event that resulted in a generator load reject from 98% of uprated power in the summer of 1999. As noted in SNOC's LER 1999-005, no anomalies were seen in the plant's response to this event. In addition, Hatch Unit 1 has experienced one turbine trip and one generator load reject event subsequent to its uprate (i.e., LERs 2000-004 and 2001-002). Again, the behavior of the primary safety systems was as expected. No new plant behaviors were observed that would indicate that the analytical models being used are not capable of modeling plant behavior at EPU conditions.

The KKL power uprate implementation program was performed during the period from 1995 to 2000. Power was raised in steps from its previous operating power level of 3138 MWt (i.e., 104.2% of OLTP) to 3515 MWt (i.e., 116.7% OLTP). Uprate testing was performed at 3327 MWt (i.e., 110.5% OLTP) in 1998, 3420 MWt (i.e., 113.5% OLTP) in 1999 and 3515 MWt in 2000.

KKL testing for major transients involved turbine trips at 110.5% OLTP and 113.5% OLTP and a generator load rejection test at 104.2% OLTP. The KKL turbine and generator trip testing demonstrated the performance of equipment that was modified in preparation for the higher power levels. Equipment that was not modified performed as before. The reactor vessel pressure was controlled at the same operating point for all of the uprated power conditions. No unexpected performance was observed except in the fine-tuning of the turbine bypass opening that was done as the series of tests progressed. These large transient tests at KKL demonstrated the response of the equipment and the reactor response. The close matches observed with

predicted response provide additional confidence that the uprate licensing analyses consistently reflected the behavior of the plant.

Plant Modeling, Data Collection, and Analyses

From the power uprate experience discussed above, it can be concluded that large transients, either planned or unplanned, have not provided any significant new information about transient modeling or actual plant response. Since the VYNPS uprate does not involve reactor pressure changes, this experience is considered applicable.

The safety analyses performed for VYNPS used the NRC-approved ODYN transient modeling code. The NRC accepts this code for GE BWRs with a range of power levels and power densities that bound the requested power uprate for VYNPS. The ODYN code has been benchmarked against BWR test data and has incorporated industry experience gained from previous transient modeling codes. ODYN uses plant specific inputs and models all the essential physical phenomena for predicting integrated plant response to the analyzed transients. Thus, the ODYN code will accurately and/or conservatively predict the integrated plant response to these transients at CPPU power levels and no new information about transient modeling is expected to be gained from performing these large transient tests.

CONCLUSION

VYNPS believes that sufficient justification has been provided to demonstrate that an MSIV transient test and a generator load rejection test is not necessary or prudent. Also, the risk imposed by intentionally initiating large transient testing should not be incurred unnecessarily. As such, Entergy does not plan to perform additional large transient testing following the VYNPS CPPU.

Docket No. 50-271
BVY 03-80

Attachment 8

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 263

Extended Power Uprate

VYNPS Environmental Assessment Report for EPU

Vermont Yankee Nuclear Power Station Environmental Assessment Report for Extended Power Uprate

September 2003

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1.0 INTRODUCTION

This Environmental Assessment Report is provided by Vermont Yankee (VY) (Reference 1) pursuant to 10 CFR 51.41, "Requirement to Submit Environmental Information," and supports the proposed change to the authorized operating power level at Vermont Yankee Nuclear Power Station (VYNPS). This evaluation provides information necessary to determine the environmental impact of those particular changes associated with the proposed power uprate at VYNPS from 1593 megawatts-thermal (MWt) to 1912 MWt. Environmental report general requirements are outlined in 10 CFR 51.45, "Environmental Report."

The environmental impact of operation at Current Licensed Thermal Power (CLTP) has been reviewed and determined to be acceptable by the Nuclear Regulatory Commission (NRC). In 1970 an Environmental Report was submitted to the Atomic Energy Commission (AEC) as part of the application for an operating license for VYNPS. The report addressed the environmental impacts of construction and operation of VYNPS and was utilized by the AEC in preparing a Final Environmental Statement (FES) (Reference 2). The AEC subsequently issued an operating license to VY authorizing operation up to a maximum power level of 1593 MWt.

This evaluation demonstrates that the proposed VYNPS power uprate to 1912 MWt will not result in a significant increase in the environmental impacts of operation of the VYNPS. The environmental impacts of VYNPS operation with extended power uprate (EPU) continue to be bounded by the FES or bounded by other appropriate regulatory criteria. This evaluation is submitted, in part, to fulfill the NRC requirement to submit a 'Supplement to the Applicant's Environmental Report' as documented in the Staff Position concerning General Electric (GE) Boiling Water Reactor (BWR) EPU Program dated February 8, 1996 (Reference 3).

This environmental report will assess the impact of EPU on the environment, compare changes to those presented in the FES or in more recent environmental reports, identify reasonable alternatives to the proposed EPU, and recommend the proper course of action.

2.0 OVERVIEW OF OPERATIONAL AND EQUIPMENT CHANGES

VYNPS is a Boiling Water Reactor (BWR) that operates in a direct thermodynamic cycle between the reactor and the turbine. Under power uprate conditions, thermodynamic processes are changed to extract additional work from the turbine. Simply put, power uprate involves an increase in the heat output of the reactor to support increased turbine inlet steam flow requirements and an increase in the heat dissipated by the condenser. No increase in reactor operating pressure or core flow is necessary to support power uprate. In the turbine portion of the heat cycle, an increase in the turbine inlet steam flow will result in an approximately proportional increase in the heat rejected either by the cooling towers or to the river, or some

combination of both. The environmental impacts of these operational changes are discussed herein.

Due to design and safety margins inherent in plant equipment, the proposed power uprate can be accomplished with relatively few modifications. The most significant changes involve replacement of the high pressure turbine steam path, rewind of the main generator, replacement of four high pressure heaters, and replacement of the main transformer (completed during 2002). A complete list of major modifications is included in the List of Planned Modifications and Associated Testing (attachment 3 of this License Amendment Request).

The modifications are being accomplished by standard maintenance and modification processes that are similar to those performed during normal outages.

3.0 PROPOSED ACTION AND NEED

3.1. Proposed Action

The proposed action is an amendment to the VY Operating License to increase the licensed core thermal power level to 1912 MWt. The operational goal of this amendment is to increase electrical generating capacity. In conjunction with the plant Nuclear Steam Supply System designer, General Electric, the effects of a power uprate at VYNPS have been comprehensively evaluated. This evaluation concluded that sufficient safety and design margins exist such that an increase in the rated core thermal power from 1593 to 1912 MWt can be accomplished without adverse impact on the health and safety of the public and without significant impact on the environment.

The unit would increase to between 110 and 115 percent of CLTP upon receipt of uprated license amendment following the spring 2004 Refueling Outage, and up to 120 percent of CLTP following the fall 2005 Refueling Outage. This supplemental Environmental Report evaluates environmental impacts of the increase to 120 percent of CLTP.

3.2. Need for Proposed Action

EPU will have the potential to provide additional electricity to Vermont and the New England area at a price that is expected to be lower than market cost. The additional 100 to 110 megawatts-electric (MWe) generated by VYNPS will be enough electricity to power approximately 110,000 homes. EPU will minimize the need to seek electricity needs elsewhere (fossil fuels), resulting in cleaner emissions. Both Vermont and other New England consumers will benefit from the proposed uprate.

A. Benefits: The State of Vermont and the Town of Vernon are likely to receive direct monetary benefits in the form of additional tax collections, to

the extent that tax collections are based on the market value of the plant or actual generation. The implementation of the proposed uprate will not decrease the value of the plant in any way. Indirect monetary benefits accruing to the state and its residents include the potential for lower electricity costs as a result of the uprate's increasing the overall regional electric supply. Stable electric costs will benefit Vermont businesses, which will consequently enhance overall economic growth in the state and increase sales and income tax collections. Broader non-monetary (social and environmental) benefits will include: (1) additional price stability for the state's ratepayers as dependence on fossil fuel generation and exposure to volatile fossil fuel markets is reduced; (2) reduced air pollution emissions, including reduced emissions of greenhouse gases to the extent that the power generated by the uprate displaces fossil-fuel generating resources; and (3) additional in-state energy supplies.

B. Costs: No direct monetary investment will be required by the State of Vermont for the proposed uprate. Entergy will bear the capital and operating costs and all of the financial risk associated with the uprate. The proposed uprate will not adversely affect the development of cost-effective demand-side management or renewable energy resources, nor impose any undue non-monetary costs on the State and its residents.

C. Conclusion: Given the lack of monetary and non-monetary costs that will be borne by the State and its residents, the tangible benefits likely to be realized, the proposed uprate will provide net economic and environmental benefits to the State and promote the general good.

ENVIRONMENTAL STATEMENT

4.0 Probable Environmental Impact

4.1. General

VYNPS impacts the environment in the following ways listed below. All items except 5 and 6 are common to any large thermal power project.

1. Discharge of large quantities of warm condenser cooling water;
2. Discharge of some permitted chemicals into the water;
3. Physical presence; i.e., structure, sounds;
4. Land use;
5. Release of some radioactive matter to the air and water; and
6. Atmospheric effects of cooling towers.

The relationship of EPU at VYNPS to its environment in terms of the above and its conformance to all applicable federal, state, and local standards in these areas are the subject of the following statements.

4.2. Thermal Effects

At extended power uprated conditions, the heat rejected to the condenser increases, resulting in an increase in the circulating water outlet temperature. In any case, VYNPS will continue to be operated according to the established cooling water discharge limitations specified in the National Pollutant Discharge Elimination System (NPDES) permit. However, VY has proposed, as a matter separate from this uprate, an amendment to the NPDES permit, to allow a one (1) degree Fahrenheit increase from the period of May 16 through October 14 (hereinafter referred to as the "NPDES summer period") water discharge temperature limitations. The NPDES Permit amendment is not required to achieve the power uprate. If the requested amendment to the NPDES permit is not granted, VYNPS will continue to operate under the current thermal discharge limits.

Operation of the cooling towers is not presently required from the period of October 15 through May 15 (hereinafter referred to as the "NPDES winter period"). The proposed power uprate may require limited operation of the cooling towers during the winter to comply with the NPDES permit. VY is considering proposing an amendment to the NPDES permit, as a matter separate from this uprate, that would not require cooling tower operation during the winter period.

4.3. Radiological Effects

It is projected that EPU will slightly increase radiation levels at the site boundary due to gaseous releases and direct radiation from the plant and stored material. It is also expected that the zero discharge of liquid effluents operating practice, currently in place, will not be impacted by uprate. VYNPS will stay within the limits of both State and Federal regulations.

<u>Direct Radiation:</u>

Normal operation off-site direct radiation doses will increase as a result of EPU. The activity level in reactor steam following EPU will increase approximately in proportion to the increase in reactor thermal power. The EPU increase in steam flow will further increase the level of activity at the main turbine. The increased flow rate and velocity, which result in shorter travel times to the turbine and less radioactive decay in transit, lead to higher radiation levels in and around the turbine and offsite dose. Based on currently reported pre-EPU doses, the maximum site boundary annual dose due to direct and skyshine from all plant sources post-EPU is estimated to be 18.6 mrem and, therefore, will remain within the state regulatory limit of 20 mrem/year and the federal regulatory limit of 25 mrem/year. This represents an increase of 3.6 mrem/year above existing site boundary dose levels.

Liquid Radwaste:

Extended power uprate will slightly increase the activity level of radioactive isotopes in the reactor coolant and steam. Due to leakage or process operations, fractions of these fluids are transported to the liquid and gaseous radwaste systems. As the activity levels in the reactor coolant and steam are increased, the activity level of radwaste inputs are proportionately increased. Liquids from reactor process systems, or liquids that have become contaminated with these process system liquids, are considered liquid radioactive waste. These wastes are then processed according to their purity level (conductivity, insoluble solids content, organic content, and activity) before being recycled within the plant as condensate, reprocessed through the radioactive waste system for further purification, or discharged to the environment as liquid radwaste effluents in accordance with federal and state discharge regulations.

Extended Power Uprate is projected to increase the processed volume of liquid radwaste by 1.2% of the current total. The total liquid radwaste volume increase is due to the increased frequency of reactor water cleanup filter demineralizer (RWCU F/D) and condensate demineralizer backwashes. The percentage increase corresponding to the RWCU F/D backwash is assumed to be proportional to the RWCU System conductivity increase. The percentage increase corresponding to the condensate F/D backwash is assumed to be proportional to the reactor feedwater flow increase. This percentage increase in liquid radwaste due to EPU conditions was evaluated and determined to be within the designed system total volume capacity.

Gaseous Radwaste:

As mentioned above, the power uprate is projected to result in a small increase in the activity level of radioactive isotopes, and in turn create more gaseous radioactive waste. Airborne particulates and gases vented from process equipment, and the building ventilation exhaust air are considered gaseous radioactive waste. The major source of gaseous radioactive waste (condenser air ejector effluent) is continuously decayed using delay pipes and charcoal adsorber beds, and filtered. Gaseous radwaste effluents are monitored prior to release to the environment to ensure that the dose guidelines of federal and state regulations are not exceeded.

Gaseous radioactive effluents are expected to increase following Extended Power Uprate but will remain well within the limits established in both state and federal regulations.

Solid Radwaste:

Extended power uprate is projected to increase the generation of solid processed radwaste at VY by 17.8% of the current total. The total solid radwaste increase from sludge and resin solids is due to the solid waste production increase from the RWCU F/D resin replacement, the condensate demineralizers resin replacement, the waste demineralizer and waste control filter resins, and the liquid radwaste handling. The percentage increase corresponding to waste demineralizer and waste control filter resins is assumed to be proportional to the increase in liquid inputs due to EPU. The percentage increase in solid radwaste due to EPU conditions was evaluated and determined to be within the designed system total volume capacity.

Conclusions:

Gaseous effluents and direct radiation dose at the site boundary will increase slightly but will continue to remain within state and federal limits. Liquid effluents are expected to stay within the zero-discharge policy that Vermont Yankee has implemented.

EPU does not have an adverse effect on the processing of liquid and solid radwaste, and there are no significant environmental effects. The increases in the liquid and the solid processed waste are primarily due to the feedwater flow increase.

EPU does not change flows, temperatures, or pressures in any portion of the liquid and solid radwaste management system. Therefore, the individual radwaste system components are not subjected to any change due to EPU.

4.4. Atmospheric Effects

VYNPS does not produce typical greenhouse gases or any noxious odors as part of the electricity production process. The operations and maintenance activities required after the uprate will be essentially the same as at present; therefore no increases in emissions or noxious odors are anticipated from them. VYNPS is in compliance with Vermont Air Pollution Control regulations as reported in its annual renewal of Air Source Registration that demonstrates that VY Station's air emissions are less than the limit of 10 tons per year. No changes are proposed for the power uprate that would increase air emissions; therefore no undue air pollution, as judged by Vermont's Air Pollution Control Regulations, will occur. The power uprate will not affect the facility's status as a Registered Source.

4.5. Chemical Release Effects

The chemicals and concentrations released into the Connecticut River as a result of EPU will comply with the NPDES permit.

Hazardous waste generation will not increase significantly following implementation of the power uprate. Typical components of the hazardous waste stream are paints and solvents used for facility maintenance. VYNPS maintains registration as a Large Quantity Generator under the Vermont Hazardous Waste Management Regulations, Section 7-308, and the VYNPS status as a Large Quantity Generator will be maintained. Asbestos removal will be required during the power uprate. The asbestos covered stator bars in the Main Generator will be removed and replaced. Four asbestos-insulated feedwater heaters that are painted with lead-based paint will be replaced during the power uprate project. All asbestos and lead abatement will be completed by certified asbestos and lead-based paint abatement contractors. All asbestos containing wastes will be disposed of at a landfill licensed for asbestos waste. The estimated quantity of asbestos to be removed is 600 cubic feet.

4.6. Physical Presence

There will be no external changes to the Vermont Yankee facility that are visible except for the replacement Main Transformer, that has slightly different dimensions than the transformer it replaced, and temporary facilities.

The cooling tower plume dimensions will increase. During the NPDES summer period (May 16 – October 14), the increase in typical cooling tower visible plume dimensions from the existing conditions to the EPU conditions is generally 100 meters in plume length and 20 to 30 meters in plume height and width, with plume height increases by as much as 50 meters (Reference 5). During the NPDES winter period, if the cooling towers are used, the EPU will result in cooling tower operation with a visible plume. Since cooling tower heat rejection rates during the NPDES winter period are lower than other times of the year, visible plume sizes during this period will be no larger than during the remainder of the year.

The Seasonal/Annual Cooling Tower Impact Program (SACTIP) evaluation shows that there are no occurrences of ground-level fogging or icing predicted in the summer under NPDES permit summer period limits and no occurrences in winter due to the low heat rejection rates under NPDES permit winter period limits. (Reference 5) Likewise, the SACTIP evaluation of fogging and icing impacts in the spring and fall seasons, which are comprised of a combination of both NPDES summer and winter periods, indicates no occurrences of icing and only a maximum of 3 hours of fogging over 5 years. (Reference 5)

The estimated increased size of the vapor plume will cause an aesthetic impact, but this impact will not be undue because of the following reasons: (1) similar plumes have been a frequent presence in the landscape since 1972, (2) the site has been an industrial facility since 1972, (3) industrial plumes are a common feature of the Connecticut River Valley, (4) the changes to the plumes do not violate a clear community standard, and (5) the plumes do not violate the sensibilities of the average person.

A noise study was performed on the cooling tower at VYNPS which concluded that, under any scenario, the cooling tower modifications associated with the EPU will result in a sound level increase of less than one decibel, which should not be noticeable. No significant increase in ambient noise levels is expected within the plant. This includes the upgraded turbine, which will operate at the same speed as the original equipment.

Silt accumulates in a basin under the west cooling tower. The basin is cleaned out periodically, and the silt is spread on the same fields that are used for the septic systems waste. The primary source of the silt is suspended solids in the cooling water taken from the river. Power uprate may cause an increase in the annual silt deposits due to the fact that more river water passes through the cooling towers, especially if the cooling towers are operated during the NPDES winter period. (Reference 6)

4.7. Land Use

Extended power uprate will not disturb the habitat of any terrestrial plant or animal species. Although occurrences of rare and threatened species and unique natural areas are located near VYNPS, the Vermont Nongame and Natural Heritage Program (VNNHP), associated with the Vermont Department of Natural Resources, reviewed the project and did not find that any undue adverse impacts would occur to nongame resources or significant natural areas. Although not tracked by the VNNHP, a bald eagle nest has been reported downstream of VYNPS on Stebbins Island, which is in New Hampshire. The uprate project will not impact this federally protected threatened species because compliance with the conditions of the current NPDES permit or any subsequent amendments will assure the protection of habitat values important for this species.

The only exterior construction required for the power uprate was the installation of temporary office space using mobile modular units. Soil disturbance was minor and limited to trenching, setting of foundation columns, hook-up of water, sewer, telephone, and electricity. Extended power uprate will not affect lands outside of the VYNPS property. Also, EPU will not involve any significant changes to aesthetic resources and does not affect any archaeological or historic sites.

4.8. Impact on the Connecticut River

The only impact on the Connecticut River under the existing NPDES permit is a slight increase in river water consumption caused by cooling tower evaporative losses and drift. During the NPDES summer period (May 16 – October 14), the increased water consumption will be less than 0.1% of average monthly river flow. During the NPDES winter period (October 15 – May 15), the increased water consumption will be less than 0.2% of average monthly river flow (Reference 4). Although not required to support EPU, the proposed NPDES amendment would reduce evaporative and drift losses to levels equal to, or less than, pre-uprate values.

VY performs continuous monitoring of river temperature and flow upstream and downstream of the facility, and also conducts water quality sampling, and ecological studies of macroinvertebrates, larval fish, and fish in the river. The purpose of this monitoring is to assure that the discharges from the VYNPS do not have an adverse impact on the fish and other wildlife communities in the river, and that the biological integrity of the aquatic community in the river is maintained. For over 30 years, the data presented in the annual monitoring reports have been reviewed by the Environmental Advisory Committee (EAC). The EAC was established by State regulators and is formally recognized as an advisory committee in the NPDES permit. The EAC members represent the Vermont Department of Environmental Conservation, the Vermont Department of Fish and Wildlife, the New Hampshire Department of Environmental Services, the New Hampshire Department of Fish and Game, the Massachusetts Department of Environmental Protection, the Massachusetts Division of Fish and Wildlife, and the United States Fish and Wildlife Service. The annual monitoring reports are also reviewed by the Vermont Agency of Natural Resources which has the regulatory authority over the NPDES permit. As monitoring is a component of the current NPDES permit, it is expected to continue during subsequent renewals or amendments.

For more than 30 years, VY has monitored fish habitat in the vicinity of the VYNPS. Seven representative important species have been selected for detailed study: Atlantic salmon, American shad, smallmouth bass, walleye, yellow perch, white perch, and spottail shiner. Monitoring to date confirms that habitat for these species has been adequately protected by the NPDES permit. Compliance with the conditions of the current NPDES permit or any subsequent amendments will assure the protection of habitat standards in the Connecticut River following implementation of the EPU.

4.9. Wastewater Discharges

There are four sources of wastewater discharge from the VYNPS. Wastewater that is discharged to the Connecticut River consists of three waste streams: (1)

the main condenser cooling water discharge; (2) the service water system discharge; and (3) stormwater discharge. The fourth source is domestic sewage from the facility that is discharged to septic systems. The main condenser cooling water and service water are a non-contact by-product from the generation of electricity. Non-contact means that the water discharged to the Connecticut River is never in direct contact with the reactor or any radioactive water. Each of the mentioned discharges is governed by conditions in the NPDES or Indirect Discharge Permits (IDP). The power uprate project will not exceed any of the permit established limits, and no amendments to existing permits are needed.

5.0 ADVERSE ENVIRONMENTAL EFFECTS

The federal and state regulations applicable to the VYNPS are designed to protect the environment, and VY will comply with these regulations. VY expects that the EPU will have no adverse environmental effects. However, if such unforeseen adverse effects, due to the EPU, were to become evident, appropriate actions as determined by VY and regulatory agencies would be taken to correct the situation.

6.0 ALTERNATIVES TO THE PROPOSED ACTION

The analysis to support the proposed uprate is essentially complete and initial implementation is scheduled for the third quarter of 2004. At this time Entergy has no alternatives to the proposed action. EPU will provide power that is necessary for Vermont and the New England area with minimum impact on its environment. Without the EPU, other agencies and electric power organizations may be required to pursue other alternate sources of power to meet the regional electric power demands. Additional generation and/or transmission capacity will be necessary to offset future demand for electrical power.

A quantitative study of environmental cost of alternatives would not be necessary to recognize that significant environmental benefits may be derived from an EPU when compared to other options regarding additional capacity. As demonstrated herein, an EPU would not result in significant environmental costs. Unlike fossil fuel plants, VYNPS does not routinely emit sulfur dioxide, nitrogen oxides, carbon dioxide, or other atmospheric pollutants during normal operation. Routine operation of VYNPS at EPU conditions would not contribute to greenhouse gases or acid rain. The radiological effects of the uranium fuel cycle are described in 10 CFR 51.51 and 10 CFR 51.52 and are classified as small. The tables depicting radiological effects in 10 CFR 51.52 encompass the EPU level. While the project would produce additional spent nuclear fuel, the added amount would not be appreciable and would be accommodated by the spent fuel management processes.

Based upon the discussion above, it is reasonable to conclude the VYNPS EPU provides an economic and environmental advantage over other alternatives for supplying additional generation. EPU involves a cost-effective utilization of an

existing asset, with relatively little environmental impact, making it the preferred means of securing additional generating capacity.

7.0 RELATIONSHIP BETWEEN SHORT TERM USES OF ENVIRONMENT AND LONG TERM PRODUCTIVITY

Maintaining the long-term productivity of the State of Vermont and New England requires that an adequate supply of electricity be generated for both industrial and private use. EPU will therefore enhance long-term productivity with minimum environmental impact.

8.0 IRRETRIEVABLE AND IRREVERSIBLE COMMITMENT OF RESOURCES

The only two resources for which there is an associated irretrievable and irreversible commitment are water and uranium fuel.

Water:

Water sources for the VYNPS are the Connecticut River and on-site groundwater wells. The quantity of groundwater used for operation under EPU will not change as a result of the power uprate. There will be a slight increase in cooling tower evaporative losses and drift as described in Section 4.2.

Uranium Fuel:

Under extended power uprate conditions, the number of fuel assemblies consumed each cycle is expected to increase by up to 28%. To support power uprate, the U-235 enrichment levels will also increase, but still be less than that assumed in VYNPS safety analyses. Although some radionuclide inventory levels and activity levels are projected to increase, little or no increase in the amount of radionuclides released to the environment during normal operation is expected (See Section 4). Incremental environmental effects of increased enrichment and burnup on transportation of fuel, spent fuel, and waste are not significant (Reference 7). In addition, there are salient environmental benefits of extended burnup, such as reduced occupational dose, reduced public dose, reduced fuel requirements per unit electricity, and reduced shipments (Reference 7).

EPU will result in additional spent nuclear fuel storage at VYNPS. However, the additional spent fuel waste will not have an unduly adverse environmental impact. Given the projected discharge rate of spent fuel for the next four fuel cycles, VYNPS can operate to the Fall 2008 refueling outage before exhausting its full-core discharge capability and reaching the capacity of the spent fuel pool. With the implementation of a full 20% power uprate, VY anticipates that the VYNPS would exhaust its full core discharge capability one cycle earlier or by Spring 2007.

With or without power uprate, VY intends to seek all necessary approvals for temporary dry fuel storage at the VYNPS within the next few years. Implementation of dry fuel storage will be necessary, even without the uprate, to allow the VYNPS to operate through its current license, which expires in 2012. The additional spent fuel resulting from power uprate has no impact on the need for dry fuel storage other than accelerating the time by which such capability is necessary by one fueling cycle.

Land and materials of construction were discussed earlier in the report. Of these, only the fissionable portion of the nuclear fuel may truly be considered irretrievably lost due to the EPU.

9.0 Conclusion

In accordance with 10CFR50.92, VY has concluded that the proposed action to increase the licensed power output of the VYNPS to 1912 MWt (i) does not involve a significant hazards consideration, (ii) there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite, and (iii) there is no significant increase in individual or cumulative occupational radiation exposure. Based on these determinations, the proposed action will not have a significant effect on the quality of the human environment.

10. References

1. Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. and the licenses of the Vermont Yankee Nuclear Power Station.
2. "Final Environmental Statement Related to the Operation of Vermont Yankee Nuclear Power Station," Docket No. 50-271, United States Atomic Energy Commission, July 1972.
3. Staff Position concerning the GE BWR EPU Program (TAC No. M91680), February 8, 1996.
4. "Extended Power Uprate Cooling Tower Evaporative Loss Study for Vermont Yankee Nuclear Power Station," Stone & Webster, Inc., Revision 1 June 2003.
5. "Extended Power Uprate Cooling Tower Visible Plume Study for Vermont Yankee Nuclear Power Station," Stone & Webster, Inc., Revision 1 June 2003.
6. "Extended Power Uprate Cooling Tower Silt Deposition Study for Vermont Yankee Nuclear Power Station," Stone & Webster, Inc., Revision 1 June 2003.
7. NUREG/CR-6703, "Environmental Effects of Extending Fuel Burnup Above 60 GWd/MTU," U.S. Nuclear Regulatory Commission, January 2001.

Docket No. 50-271
BVY 03-80

Attachment 9

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 263

Extended Power Uprate

Marked-up Technical Specification Pages

E. Entergy Nuclear Operations, Inc., pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not to separate, such byproduct and special nuclear material as may be produced by operation of the facility.

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

A. Maximum Power Level

1912 1

Entergy Nuclear Operations, Inc. is authorized to operate the facility at reactor core power levels not to exceed 1593 megawatts thermal in accordance with the Technical Specifications (Appendix A) appended hereto.

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment 208, are hereby incorporated in the license. Entergy Nuclear Operations, Inc. shall operate the facility in accordance with the Technical Specifications.

C. Reports

Entergy Nuclear Operations, Inc. shall make reports in accordance with the requirements of the Technical Specifications.

D. Records

Entergy Nuclear Operations, Inc. shall keep facility operating records in accordance with the requirements of the Technical Specifications.

E. Environmental Conditions

Pursuant to the Initial Decision of the presiding Atomic Safety and Licensing Board issued February 27, 1973, the following conditions for the protection of the environment are incorporated herein:

or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system or the coincident tripping of two trip systems.

3. Protective Action - An action initiated by the protection system when a limit is reached. A protective action can be at a channel or system level.

4. Protective Function - A system protective action which results from the protective action of the channels monitoring a particular plant condition.

2

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P. Rated Neutron Flux - Rated neutron flux is the neutron flux that corresponds to a steady state power level of ~~1593~~ thermal megawatts.

Q. Rated Thermal Power - Rated thermal power means a steady state power level of ~~1593~~ thermal megawatts.

R. Reactor Power Operation - Reactor power operation is any operation with the mode switch in the "Startup/Hot Standby" or "Run" position with the reactor critical and above 1% rated thermal power.

1. Startup/Hot Standby Mode - In this mode the low turbine condenser vacuum trip is bypassed when condenser vacuum is less than 12 inches Hg and both turbine stop valves and bypass valves are closed; the low pressure and the 10 percent closure main steamline isolation valve closure trips are bypassed; the reactor protection system is energized with IRM neutron monitoring system trips and control rod withdrawal interlocks in service and APRM neutron monitoring system operable.

2. Run Mode - In this mode the reactor system pressure is equal to or greater than 800 psig and the reactor protection system is energized with APRM protection and RBM interlocks in service.

S. Reactor Vessel Pressure - Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space detector.

T. Refueling Outage - Refueling outage is the period of time between the shutdown of the unit prior to a refueling and the startup of the plant subsequent to that refueling. For the purpose of designating frequency of testing and surveillance, a refueling outage shall mean a regularly scheduled refueling outage; however, where such outages occur within 8 months of the completion of the previous refueling outage, the required surveillance testing need not be performed until the next regularly scheduled outage.

U. Deleted

1.1 SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

Applicability:

Applies to the interrelated variable associated with fuel thermal behavior.

Objective:

To establish limits below which the integrity of the fuel cladding is preserved.

Specification:

A. Bundle Safety Limit (Reactor Pressure >800 psia and Core Flow >10% of Rated)

When the reactor pressure is >800 psia and the core flow is greater than 10% of rated:

1. A Minimum Critical Power Ratio (MCPR) of less than 1.10 (1.12 for Single Loop Operation) shall constitute violation of the Fuel Cladding Integrity Safety Limit (FCISL).

<INSERT #1>

3

2.1 LIMITING SAFETY SYSTEM SETTING

2.1 FUEL CLADDING INTEGRITY

Applicability:

Applies to trip setting of the instruments and devices which are provided to prevent the nuclear system safety limits from being exceeded.

Objective:

To define the level of the process variable at which automatic protective action is initiated.

Specification:

A. Trip Settings

The limiting safety system trip settings shall be as specified below:

1. Neutron Flux Trip Settings

a. APRM Flux Scram Trip Setting (Run Mode)

When the mode switch is in the RUN position, the APRM flux scram trip setting shall be as shown on Figure 2.1.1 and shall be:

Two loop operation:
$S \leq 0.4W + 64.4\%$ for $0\% < W \leq 31.1\%$
$S \leq 1.28W + 37.0\%$ for $31.1\% < W \leq 54.0\%$
$S \leq 0.66W + 70.5\%$ for $54.0\% < W \leq 75.0\%$
With a maximum of 120.0% power for W > 75.0%

Single loop operation:
$S \leq 0.4W + 61.2\%$ for $0\% < W \leq 39.1\%$
$S \leq 1.28W + 26.8\%$ for $39.1\% < W \leq 61.9\%$
$S \leq 0.66W + 65.2\%$ for $61.9\% < W \leq 83.0\%$
With a maximum of 120.0% power for W > 83.0%

where:

S = setting in percent of rated thermal power 1912 (~~1593~~ MWt)

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PC-263 INSERTS

INSERT #1

Two loop operation:

$S \leq 0.33\,W + 53.7\%$ for $0\% < W \leq 30.9\%$
$S \leq 1.07\,W + 30.8\%$ for $30.9\% < W \leq 66.7\%$
$S \leq 0.55\,W + 65.5\%$ for $66.7\% < W \leq 99.0\%$
With a maximum of 120.0% power for $W > 99.0\%$

Single loop operation:

$S \leq 0.33\,W + 51.1\%$ for $0\% < W \leq 39.1\%$
$S \leq 1.07\,W + 22.2\%$ for $39.1\% < W \leq 61.7\%$
$S \leq 0.55\,W + 54.3\%$ for $61.7\% < W \leq 119.4\%$
With a maximum of 120.0% power for $W > 119.4\%$

1.1 SAFETY LIMIT

B. Core Thermal Power Limit (Reactor Pressure $\le$ 800 psia or Core Flow $\le$ 10% of Rated)

When the reactor pressure is $\le$800 psia or core flow $\le$10% of rated, the core thermal power shall not exceed 25% of rated thermal power.

23%

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C. Power Transient

To ensure that the safety limit established in Specification 1.1A and 1.1B is not exceeded, each required scram shall be initiated by its expected scram signal. The safety limit shall be assumed to be exceeded when scram is accomplished by means other than the expected scram signal.

D. Whenever the reactor is shutdown with irradiated fuel in the reactor vessel, the water level shall not be less than 12 inches above the top of the enriched fuel when it is seated in the core.

2.1 LIMITING SAFETY SYSTEM SETTING

W = percent rated two loop drive flow where 100% rated drive flow is that flow equivalent to 48 x 10^6 lbs/hr core flow

In the event of operation at >25% Rated Thermal Power the APRM gain shall be equal to or greater than 1.0.

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1.1 SAFETY LIMIT

25%

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2.1 LIMITING SAFETY SYSTEM SETTING

D. Reactor low-low water level Emergency Core Cooling System (ECCS) initiation shall be at least 82.5 inches above the top of the enriched fuel.

E. Turbine stop valve scram shall, when operating at greater than 30% of Rated Thermal Power, be less than or equal to 10% valve closure from full open.

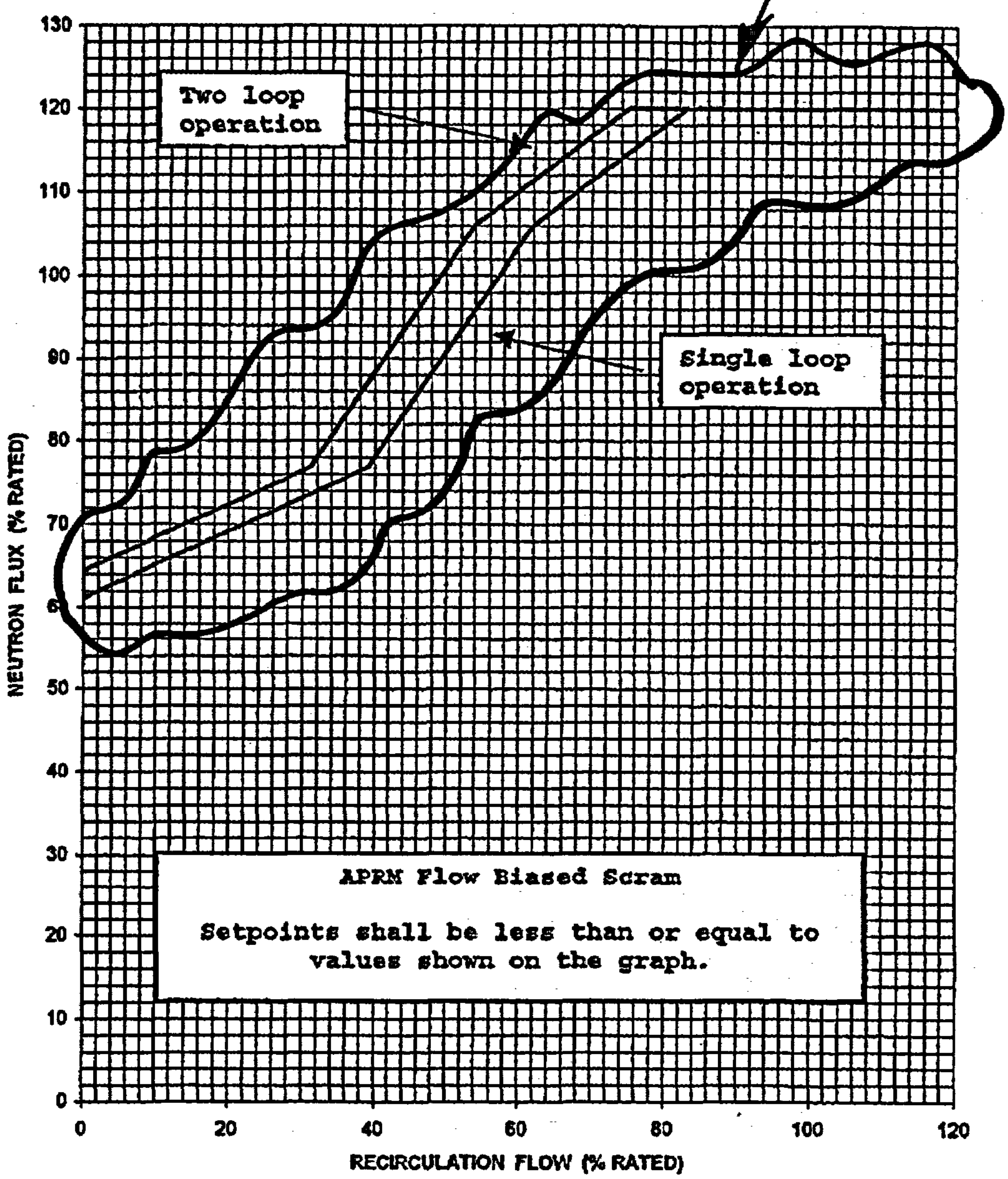
F. Turbine control valve fast closure scram shall, when operating at greater than 20% of Rated Thermal Power, trip upon actuation of the turbine control valve fast closure relay.

G. Main steam line isolation valve closure scram shall be less than or equal to 10% valve closure from full open.

H. Main steam line low pressure initiation of main steam line isolation valve closure shall be at least 800 psig.

6 REPLACE WITH INSERT #2

FIGURE 2.1.1
APRM FLOW REFERENCE SCRAM SETTING



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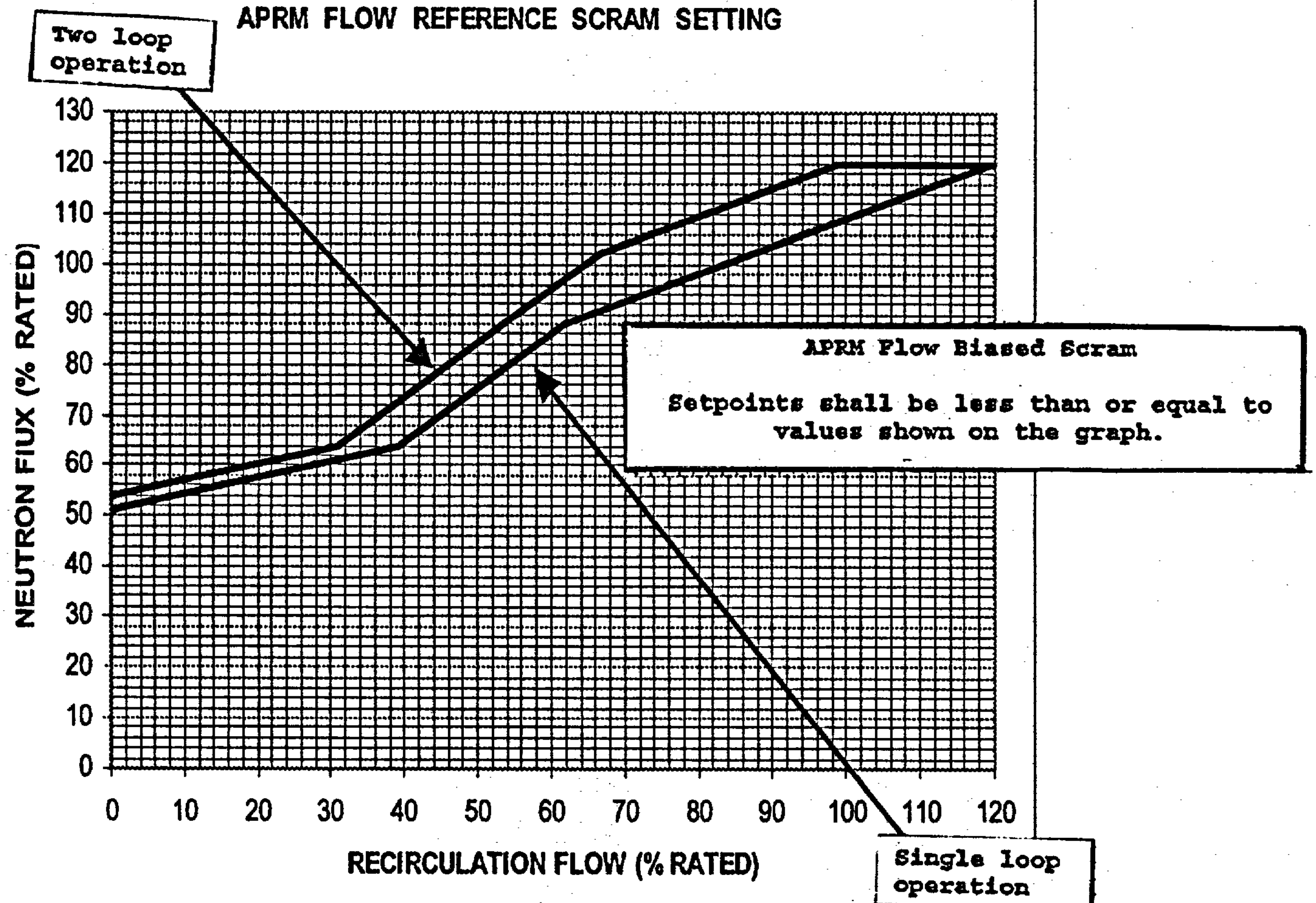
[THIS IS INSERT #2]

FIGURE 2.1-1

APRM FLOW REFERENCE SCRAM SETTING



(RE-FORMAT)

BASES:

1.1 FUEL CLADDING INTEGRITY

A. Refer to General Electric Company Licensing Topical Report, "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A (most recent revision).

The fuel cladding integrity Safety Limit (SL) is set such that no significant fuel damage is calculated to occur if the limit is not violated. Since the parameters that result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions that result in the onset of transition boiling have been used to mark the beginning of the region in which fuel damage could occur. Although it is recognized that the onset of transition boiling would not result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity SL is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition, considering the power distribution within the core and all uncertainties.

The MCPR SL is determined using a statistical model that combines all the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved General Electric Critical Power correlations.

The MCPR fuel cladding integrity SL is increased for single loop operation in order to account for increased core flow measurement and TIP reading uncertainties.

B. Core Thermal Power Limit (Reactor Pressure ≤ 800 psia or Core Flow ≤10% of Rated)

At pressures below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low power and all flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and all flows will always be greater than 4.56 psi. Analyses show that with a flow of 28×10^3 lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than 28×10^3 lbs/hr irrespective of total core flow and independent of bundle power for the range of bundle powers of concern. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors this corresponds to a core thermal power of more than 50%. Thus, a core thermal power limit of 25% for reactor pressures below 800 psia or core flow less than 10% is conservative.

23% 20

C. Power Transient

Plant safety analyses have shown that the scrams caused by exceeding any safety setting will assure that the Safety Limit of Specification 1.1.1A or 1.1.1B will not be exceeded. Scram times are checked periodically to assure the insertion times are adequate. The thermal power transient resulting when a scram is accomplished other than by the expected scram signal (e.g., scram from neutron flux

BASES:

2.1 FUEL CLADDING INTEGRITY

A. Trip Settings

The bases for individual trip settings are discussed in the following paragraphs.

1. Neutron Flux Trip Settings

a. APRM Flux Scram Trip Setting (Run Mode)

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The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady state conditions, reads in percent of rated thermal power (1593 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses are performed to demonstrate that the APRM flux scram over the range of settings from a maximum of 120% to the minimum flow biased setting provide protection from the fuel safety limit for all abnormal operational transients including those that may result in a thermal hydraulic instability.

An increase in the APRM scram trip setting would decrease the margin present before the fuel cladding integrity Safety Limit is reached. The APRM scram trip setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM scram trip setting was selected because it provides adequate margin for the fuel cladding integrity Safety Limit yet allows operating margin that reduces the possibility of unnecessary scrams. The relationship between recirculation drive flow and reactor core flow is non-linear at low core flows. Therefore, separate APRM flow biased scram trip setting equations are provided for low core flows.

The scram trip is set to ensure acceptable transient response. For single recirculation loop operation, the APRM flux scram trip setting is reduced in accordance with the analysis presented in NEDO-30060, February 1983. This adjustment accounts for the difference between the single loop and two loop drive flow at the same core flow, and ensures that the margin of safety is not reduced during single loop operation. The single loop operation equations are based on a [illegible] difference between two [illegible] flow of 8%.

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23% 20

BASES: 2.1 (Cont'd)

Flux Scram Trip Setting (Refuel or Startup and Hot Standby Mode)

For operation in the startup mode while the reactor is at low pressure, the reduced APRM scram setting to 15% of rated power provides adequate thermal margin between the setpoint and the safety limit, 25% of the rated. (During an outage when it is necessary to check refuel interlocks, the mode switch must be moved to the startup position. Since the APRM reduced scram may be inoperable at that time due to the disconnection of the LPRMs, it is required that the IRM scram and the SRM scram in noncoincidence be in effect. This will ensure that adequate thermal margin is maintained between the setpoint and the safety limit.) The margin is adequate to accommodate anticipated maneuvers associated with station startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5% of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The reduced APRM scram remains active until the mode switch is placed in the RUN position. This switch can occur when reactor pressure is greater than 800 psig. 20 psia

The IRM system consists of 6 chambers, 3 in each of the reactor protection system logic channels. The IRM is a 5-decade instrument, which covers the range of power level between that covered by the SRM and the APRM. The 5 decades are covered by the IRM by means of a range switch and the 5 decades are broken down into 10 ranges, each being one-half of a decade in size. The IRM scram trip setting of 120/125 of full scale is active in each range of the IRM. For example, if the instrument were on range 1, the scram setting would be a 120/125 of full scale for that range; likewise, if the instrument were on range 5, the scram would be 120/125 of full scale on that range. Thus, as the IRM is ranged up to accommodate the increase in power level, the scram trip setting is also ranged up. The most significant sources of reactivity change during the power increase are due to control rod withdrawal. For in-sequence control rod withdrawal, the rate of change of power is slow enough due to the physical limitation of withdrawing control rods, that heat flux is in equilibrium with the neutron flux and an IRM scram would result in a reactor shutdown well before any safety limit is exceeded.

BASES: 2.1 (Cont'd)

E. Turbine Stop Valve Closure Scram Trip Setting

The turbine stop valve closure scram trip anticipates the pressure, neutron flux and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of <10% of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above the fuel cladding integrity safety limit even during the worst case transient that assumes the turbine bypass is closed. This scram signal may be bypassed at ≤30% of reactor Rated Thermal Power.

25% 20

F. Turbine Control Valve Fast Closure Scram

The control valve fast closure scram is provided to limit the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves due to a load rejection coincident with failure of the bypass system. This transient is less severe than the turbine stop valve closure with failure of the bypass valves and therefore adequate margin exists. This scram signal may be bypassed at ≤30% of reactor Rated Thermal Power.

G. Main Steam Line Isolation Valve Closure Scram

The isolation scram anticipates the pressure and flux transients which occur during an isolation event and the loss of inventory during a pipe break. This action minimizes the effect of this event on the fuel and pressure vessel.

H. Reactor Coolant Low Pressure Initiation of Main Steam Isolation Valve Closure

The low pressure isolation of the main steam lines at 800 psig is provided to give protection against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage is taken of the scram feature which occurs when the main steam line isolation valves are closed, to provide the reactor shutdown so that high power operation at low reactor pressure does not occur. Operation of the reactor at pressures lower than 800 psig requires that the reactor mode switch be in the startup position where protection of the fuel cladding integrity safety limit is provided by the IRM high neutron flux scram.

Thus, the combination of main steam line low pressure isolation and isolation valve closure scram assures the availability of neutron scram protection over the entire range of applicability of the fuel cladding integrity safety limit.

TABLE 3.1.1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT REQUIREMENTS

	Trip Function	Trip Settings	Modes in Which Functions Must be Operating Refuel (1)	Startup (12)	Run	Minimum Number Operating Instrument Channels Per Trip System (2)	Required ACTIONS When Minimum Conditions For Operation Are Not Satisfied (3)
1.	Mode Switch in Shutdown (5A-S1)		X	X	X	1	A
2.	Manual Scram (5A-S3A/B)		X	X	X	1	A
3.	IRM (7-41(A-F))						
	High Flux	≤120/125	X	X		2	A
	INOP		X	X		2	A
4.	APRM (APRM A-F)						
	High Flux (flow bias) 7 <INSERT #1>	~~Two loop operation: (4)~~ ~~S≤ 0.4W+ 64.4% for 0% < W ≤ 31.1%~~ ~~S≤ 1.28W+ 37.0% for 31.1% < W ≤ 54.0%~~ ~~S≤ 0.66W+ 70.5% for 54.0% < W ≤ 75.0%~~ ~~With a maximum of 120.0% power for W > 75.0%~~ ~~Single loop operation: (4)~~ ~~S≤ 0.4W+ 61.2% for 0% < W ≤ 39.1%~~ ~~S≤ 1.28W+ 26.8% for 39.1% < W ≤ 61.9%~~ ~~S≤ 0.66W+ 65.2% for 61.9% < W ≤ 83.0%~~ ~~With a maximum of 120.0% power for W > 83.0%~~			X	2	A or B
	High Flux (reduced)	≤15%					
	INOP						
5.	High Reactor Pressure (PT-2-3-55(A-D)(M))	≤1055 psig					

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TABLE 3.1.1 NOTES (Cont'd)

25% 8

3. When the requirements in the column "Minimum Number of Operating Instrument Channels Per Trip System" cannot be met for one system, that system shall be tripped. If the requirements cannot be met for both trip systems, the appropriate ACTIONS listed below shall be taken:

   a) Initiate insertion of operable rods and complete insertion of all operable rods within four hours.
   b) Reduce power level to IRM range and place mode switch in the "Startup/Hot Standby" position within eight hours.
   c) Reduce turbine load and close main steam line isolation valves within 8 hours.
   d) Reduce reactor power to less than 30% of rated within 8 hours.

4. "W" is percent rated two loop drive flow where 100% rated drive flow is that flow equivalent to 48 x 10^6 lbs/hr core flow.

5. To be considered operable an APRM must have at least 2 LPRM inputs per level and at least a total of 13 LPRM inputs, except that channels A, C, D, and F may lose all LPRM inputs from the companion APRM Cabinet plus one additional LPRM input and still be considered operable.

6. The top of the enriched fuel has been designated as 0 inches and provides common reference level for all vessel water level instrumentation.

7. Deleted.

8. Deleted.

9. Channel signals for the turbine control valve fast closure trip shall be derived from the same event or events which cause the control valve fast closure.

10. Turbine stop valve closure and turbine control valve fast closure scram signals may be bypassed at ≤30% of reactor Rated Thermal Power.

25% 9

11. Not used.

12. While performing refuel interlock checks which require the mode switch to be in Startup, the reduced APRM high flux scram need not be operable provided:

    a. The following trip functions are operable:

       1. Mode switch in shutdown,
       2. Manual scram,
       3. High flux IRM scram
       4. High flux SRM scram in noncoincidence,
       5. Scram discharge volume high water level, and;

    b. No more than two (2) control rods withdrawn. The two (2) control rods that can be withdrawn cannot be face adjacent or diagonally adjacent.

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BASES: 3.1 (Cont'd)

Instrumentation is provided to detect a loss-of-coolant accident and initiate the core standby cooling equipment. This instrumentation is a backup to the water level instrumentation which is discussed in Specification 3.2.

The Control Rod Drive Scram System is designed so that all of the water that is discharged from the reactor by the scram can be accommodated in the discharge piping. This discharge piping is divided into two sections. One section services the control rod drives on the north side of the reactor, the other serves the control rod drives of the south side. A part of the piping in each section is an instrument volume which accommodates in excess of 21 gallons of water and is at the low point in the piping. No credit was taken for this volume in the design of the discharge piping as concerns the amount of water which must be accommodated during a scram. During normal operation, the discharge volume is empty; however, should it fill with water, the water discharged to the piping from the reactor could not be accommodated, which would result in slow scram times or partial or no control rod insertion. To preclude this occurrence, level instrumentation has been provided for the instrument volume which scram the reactor when the volume of water reaches 21 gallons. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharged water, and precludes the situation in which a scram would be required but not be able to perform its function adequately. The present design of the Scram Discharge System is in concert with the BWR Owner's Group criteria, which have previously been endorsed by the NRC in their generic "Safety Evaluation Report (SER) for Scram Discharge Systems", dated December 1, 1980.

Loss of condenser vacuum occurs when the condenser can no longer handle the heat input. Loss of condenser vacuum initiates a closure of the turbine stop valves and turbine bypass valves which eliminates the heat input to the condenser. Closure of the turbine stop and bypass valves causes a pressure transient, neutron flux rise, and an increase in surface heat flux. To prevent the clad safety limit from being exceeded if this occurs, a reactor scram occurs on turbine stop valve closure. The turbine stop valve closure scram function alone is adequate to prevent the clad safety limit from being exceeded in the event of a turbine trip transient without bypass.

25% — 20

Turbine stop valve (TSV) closure and turbine control valve (TCV) fast closure scram signals may be bypassed at ~~30%~~ of reactor Rated Thermal Power since, at low thermal power levels, the margins to fuel thermal-hydraulic limits and reactor primary coolant boundary pressure limits are large and an immediate scram is not necessary. This bypass function is normally accomplished automatically by pressure switches sensing turbine first stage pressure. The turbine first stage pressure setpoint controlling the bypass of the scram signals on TCV fast closure and TSV closure is derived from analysis of reactor pressurization transients. Certain operational factors, such as turbine bypass valves open, can influence the relationship between turbine first stage pressure and reactor Rated Thermal Power. However, above ~~30%~~ of reactor Rated Thermal Power, these scram functions must be enabled.

25% — 20

3.3 LIMITING CONDITIONS FOR OPERATION

2. The Control Rod Drive Housing Support System shall be in place when the Reactor Coolant System is pressurized above atmospheric pressure with fuel in the reactor vessel unless all operable control rods are fully inserted.

3. While the reactor is below 20% power, the Rod Worth Minimizer (RWM) shall be operating while moving control rods except that:

17%

10

(a) If after withdrawal of at least 12 control rods during a startup, the RWM fails, the startup may continue provided a second licensed operator verifies that the operator at the reactor console is following the control rod program; or

(b) If all rods, except those that cannot be moved with control rod drive

4.3 SURVEILLANCE REQUIREMENTS

positive coupling and the results of each test shall be recorded. The drive and blade shall be coupled and fully withdrawn. The position and over-travel lights shall be observed.

2. The Control Rod Drive Housing Support System shall be inspected after reassembly and the results of the inspection recorded.

3. Prior to control rod withdrawal for startup the Rod Worth Minimizer (RWM) shall be verified as operable by performing the following:

(a) Verify that the control rod withdrawal sequence for the Rod Worth Minimizer computer is correct.

(b) The Rod Worth Minimizer diagnostic test shall be performed.

BASES: 3.3 & 4.3 (Cont'd)

2. The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage of the primary coolant system. The design basis is given in Subsection 3.5.2 of the FSAR, and the design evaluation is given in Subsection 3.5.4. This support is not required if the reactor coolant system is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing.

3. In the course of performing normal startup and shutdown procedures, a pre-specified sequence for the withdrawal or insertion of control rods is followed. Control rod dropout accidents which might lead to significant core damage, cannot occur if this sequence of rod withdrawals or insertions is followed. The Rod Worth Minimizer restricts withdrawals and insertions to those listed in the pre-specified sequence and provides an additional check that the reactor operator is following prescribed sequence. Although beginning a reactor startup without having the RWM operable would entail unnecessary risk, continuing to withdraw rods if the RWM fails subsequently is acceptable if a second licensed operator verifies the withdrawal sequence. Continuing the startup increases core power, reduces the rod worth and reduces the consequences of dropping any rod. Withdrawal of rods for testing is permitted with the RWM inoperable, if the reactor is subcritical and all other rods are fully inserted. Above 20% power, the RWM is not needed since even with a single error an operator cannot withdraw a rod with sufficient worth, which if dropped, would result in anything but minor consequences. 20

17%

4. Refer to the "General Electric Standa[...]n for Reactor Fuel (GESTAR II)," NEDE-24011-P-A, (the la[...]roved version will be listed in the COLR).

5. The Source Range Monitor (SRM) system provides a scram function in noncoincident configuration. It does provide the operator with a visual indication of neutron level. The consequences of reactivity accidents are a function of the initial neutron flux. The requirement of at least three counts per second assures that any transient, should it occur, begins at or above the initial value of 10^{-8} of rated power used in the analyses of transients from cold conditions. One operable SRM channel is adequate to monitor the approach to criticality, therefore, two operable SRM's are specified for added conservatism.

6. Deleted.

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3.4 LIMITING CONDITIONS FOR OPERATION

3.4 REACTOR STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the operating status of the Reactor Standby Liquid Control System.

Objective:

To assure the availability of an independent reactivity control mechanism.

Specification:

A. Normal Operation

Except as specified in 3.4.B below, the Standby Liquid Control System shall be operable when the reactor mode switch is in either the "Startup/Hot Standby" or "Run" position, except to allow testing of instrumentation associated with the reactor mode switch interlock functions provided:

1. Reactor coolant temperature is less than or equal to 212° F;

2. All control rods remain fully inserted in core cells containing one or more fuel assemblies; and

3. No core alterations are in progress.

4.4 SURVEILLANCE REQUIREMENTS

4.4 REACTOR STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the periodic testing requirement for the Reactor Standby Liquid Control System.

Objective:

To verify the operability of the Standby Liquid Control System.

11

Specification:

≥ 1325

A. Normal Operation

The Standby Liquid Control System shall be verified operable by:

1. Testing pumps and valves in accordance with Specification 4.6.E. A minimum flow rate of 35 gpm at 1320 psig shall be verified for each pump.

2. Verifying the continuity of the explosive charges at least monthly.

In addition, at least once during each operating cycle, the Standby Liquid Control System shall be verified operable by:

3. Testing that the setting of the pressure relief valves is between 1400 and 1490 psig.

4. Initiating one of the standby liquid control loops, excluding the primer chamber and inlet fitting, and verifying that a flow path from a pump to the reactor vessel is available. ... loops shall be ... course ... es.

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3.4 LIMITING CONDITIONS FOR OPERATION

2. The solution temperature, including that in the pump suction piping, shall be maintained above the curve shown in Figure 3.4.2.

3. The combination of Standby Liquid Control System pump flow rate, boron concentration, and boron enrichment shall satisfy the following relationship for the Standby Liquid Control System to be considered operable:

$$\frac{Q}{86} \times \frac{M251}{M} \times \frac{C}{13} \times \frac{E}{19.8} \geq \cancel{1} \quad 1.29$$

12

where:

C = the concentration of sodium pentaborate solution (weight percent) in the Standby Liquid Control System tank

E = the boron-10 enrichment (atom percent) of the sodium pentaborate solution

~~Q = 35 gpm~~ Q = ≥ 35 gpm

$\frac{M251}{M}$ = a constant (the ratio of mass of water in the reference plant compared to VY)

D. If Specification 3.4.A or B is not met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

E. If Specification 3.4.C is not met, action shall be immediately initiated to correct the deficiency. If at the end of 12 hours the system has not been restored to full operability, then a shutdown shall be initiated with the reactor in cold shutdown within 24 hours of initial discovery.

4.4 SURVEILLANCE REQUIREMENTS

2. Sodium pentaborate concentration shall be determined at least once a month and within 24 hours following the addition of water or boron, or if the solution temperature drops below the limits specified by Figure 3.4.2.

3. The boron-10 enrichment of the borated solution required by Specification 3.4.C.3 shall be tested and verified once per operating cycle.

BASES:

3.4 & 4.4 REACTOR STANDBY LIQUID CONTROL SYSTEM

A. Normal Operation

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The desigr (SLCS) ... power withd... Standby ... which produc... reactor core in ... concentration in the reac... ...rom full power to a 5% Δk subcritical condition. An additional margin (25% of boron) is added for possible imperfect mixing of the chemical solution in the reactor water. A minimum quantity of 3850 gallons of solution having a 10.1% natural sodium pentaborate concentration is required to meet this shutdown requirement.

The time requirement (138 minutes) for insertion of the boron solution was selected to override the rate of reactivity insertion due to cooldown of the reactor following the xenon poison peak. For a required minimum pumping rate of 35 gallons per minute, the maximum net storage volume of the boron solution is established as 4830 gallons.

In addition to its original design basis, the Standby Liquid Control System also satisfies the requirements of 10CFR50.62(c)(4) on anticipated transients without scram (ATWS) by using enriched boron. The ATWS rule adds hot shutdown and neutron absorber (i.e., boron-10) injection rate requirements that exceed the original Standby Liquid Control System design basis. However, changes to the Standby Liquid Control System as a result of the ATWS rule have not invalidated the original design basis.

With the reactor mode switch in the "Run" or "Startup/Hot Standby" position, shutdown capability is required. With the mode switch in "Shutdown," control rods are not able to be withdrawn since a control rod block is applied. This provides adequate controls to ensure that the reactor remains subcritical. With the mode switch in "Refuel," only a single control rod can be withdrawn from a core cell containing fuel assemblies. Determination of adequate shutdown margin by Specification 3.3.A ensures that the reactor will not become critical. Therefore, the Standby Liquid Control System is not required to be operable when only a single control rod can be withdrawn.

1325

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1292

Pump operability testing (by recirculating demineralized water to the test tank) in accordance with Specification 4.6.E is adequate to detect if failures have occurred. Flow, relief valve, circuitry, and trigger assembly testing at the prescribed intervals assures a high reliability of system operation capability. The maximum SLCS pump discharge pressure during the limiting ATWS event is ~~1320~~ psig. This value is based on a ~~peak~~ reactor vessel lower plenum pressure of ~~1290~~ psia that occurs during the limiting ATWS event at the time of SLCS initiation, i.e., 120 seconds into the event. There is adequate margin to prevent the SLCS relief valve from lifting. With a nominal SLCS relief valve setpoint of 1400 psig, there is a margin of ~~80~~ psi between the peak SLCS pump discharge pressure and the relief valve nominal setpoint. Recirculation of the borated solution is done during each operating cycle to ensure one suction line from the boron tank is clear. In addition, at least once during each operating cycle, one of the standby liquid control loops will be initiated to verify that a flow path from a pump to the reactor vessel is available by pumping demineralized water into the reactor vessel.

75

20

BASES: 3.4 & 4.4 (Cont'd)

B. Operation With Inoperable Components

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C. Standby Liquid Control System Tank - Borated Solution

The solution saturation temperature varies with the concentration of sodium pentaborate. The solution shall be kept at least 10°F above the saturation temperature to guard against boron precipitation. The 10°F margin is included in Figure 3.4.2. Temperature and liquid level alarms for the system are annunciated in the Control Room.

Once the solution has been made up, boron concentration will not vary unless more boron or water is added. Level indication and alarm indicate whether the solution volume has changed which might indicate a possible solution concentration change. Considering these factors, the test interval has been established.

Sodium pentaborate concentration is determined within 24 hours following the addition of water or boron, or if the solution temperature drops below specified limits. The 24-hour limit allows for 8 hours of mixing, subsequent testing, and notification of shift personnel.

Boron concentration, solution temperature, and volume are checked on a frequency to assure a high reliability of operation of the system should it ever be required. Isotopic tests of the sodium pentaborate are performed periodically to ensure that the proper boron-10 atom percentage is being used.

10CFR50.62(c)(4) requires a Standby Liquid Control System with a minimum flow capacity and boron content equivalent to 86 gpm of 13 weight percent natural sodium pentaborate solution in the 251-inch reactor pressure vessel reference plant. Natural sodium pentaborate solution is 19.8 atom percent boron-10. The relationship expressed in Specification 3.4.C.3 also contains the ratio M251/M to account for the difference in water volume between the reference plant and Vermont Yankee. (This ratio of masses is 628,300 lbs./401,247 lbs.)

To comply with the ATWS rule, the combination of three Standby Liquid Control System parameters must be considered: boron concentration, Standby Liquid Control System pump flow rate, and boron-10 enrichment. ~~Fixing the pump flow rate in Specification 3.4.C.3 at the minimum flow rate of 35 gpm conservatively establishes a system parameter that can be used in satisfying the ATWS requirement as well as the original system design basis.~~ If the product of the expression in Specification 3.4.C.3 is equal to or greater than ~~unity~~ 1.29, the Standby Liquid Control System satisfies the requirements of 10CFR50.62(c)(4) and the plant-specific ATWS analysis.

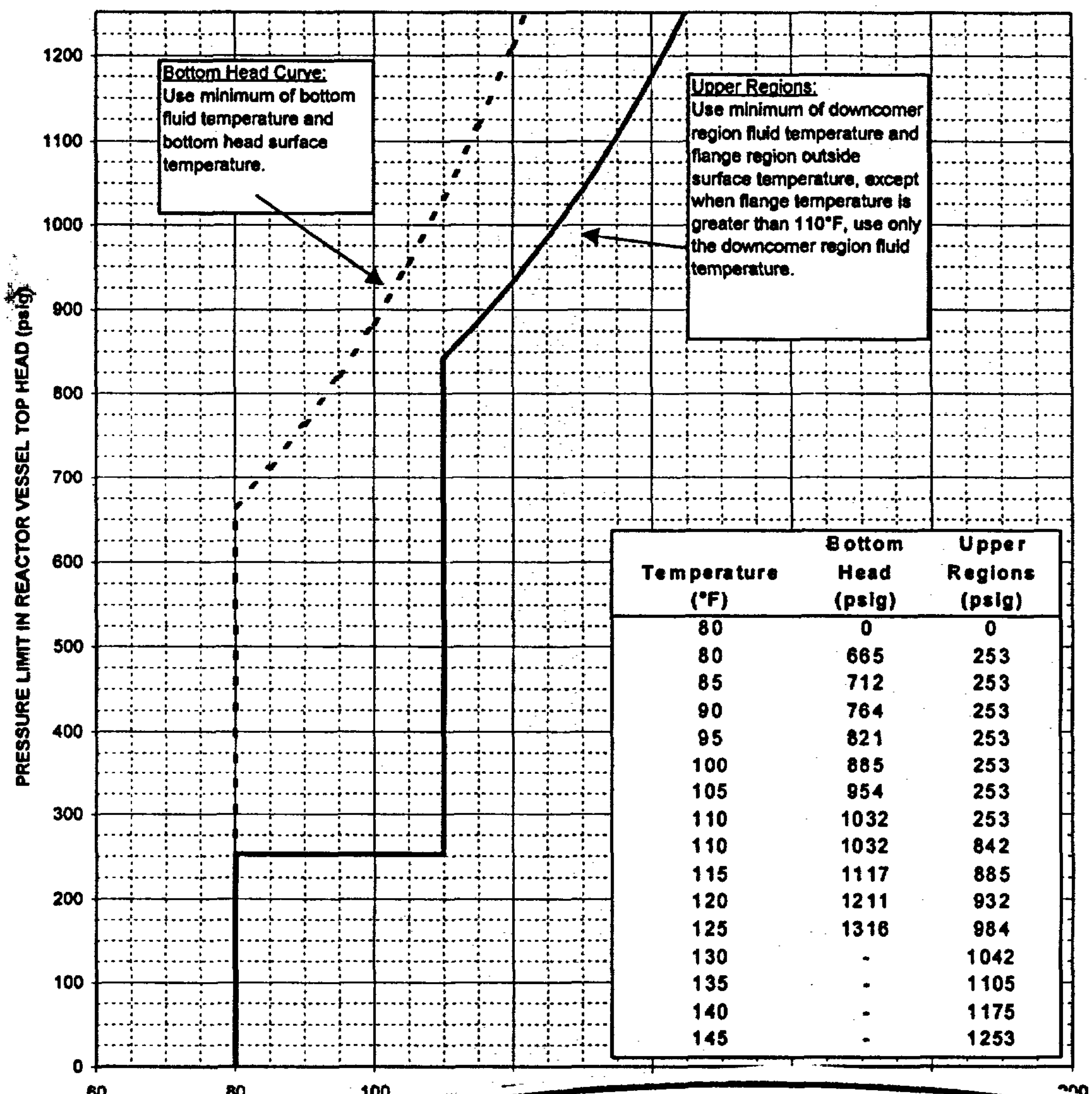
Figure 3.6.1

Reactor Vessel Pressure-Temperature Limitations
Hydrostatic Pressure and Leak Tests, Core Not Critical

40°F/hr Heatup/Cooldown Limit
Valid Through 4.46E8 MWH(t)

4.827E8

13



Temperature (°F)	Bottom Head (psig)	Upper Regions (psig)
80	0	0
80	665	253
85	712	253
90	764	253
95	821	253
100	885	253
105	954	253
110	1032	253
110	1032	842
115	1117	885
120	1211	932
125	1316	984
130	-	1042
135	-	1105
140	-	1175
145	-	1253

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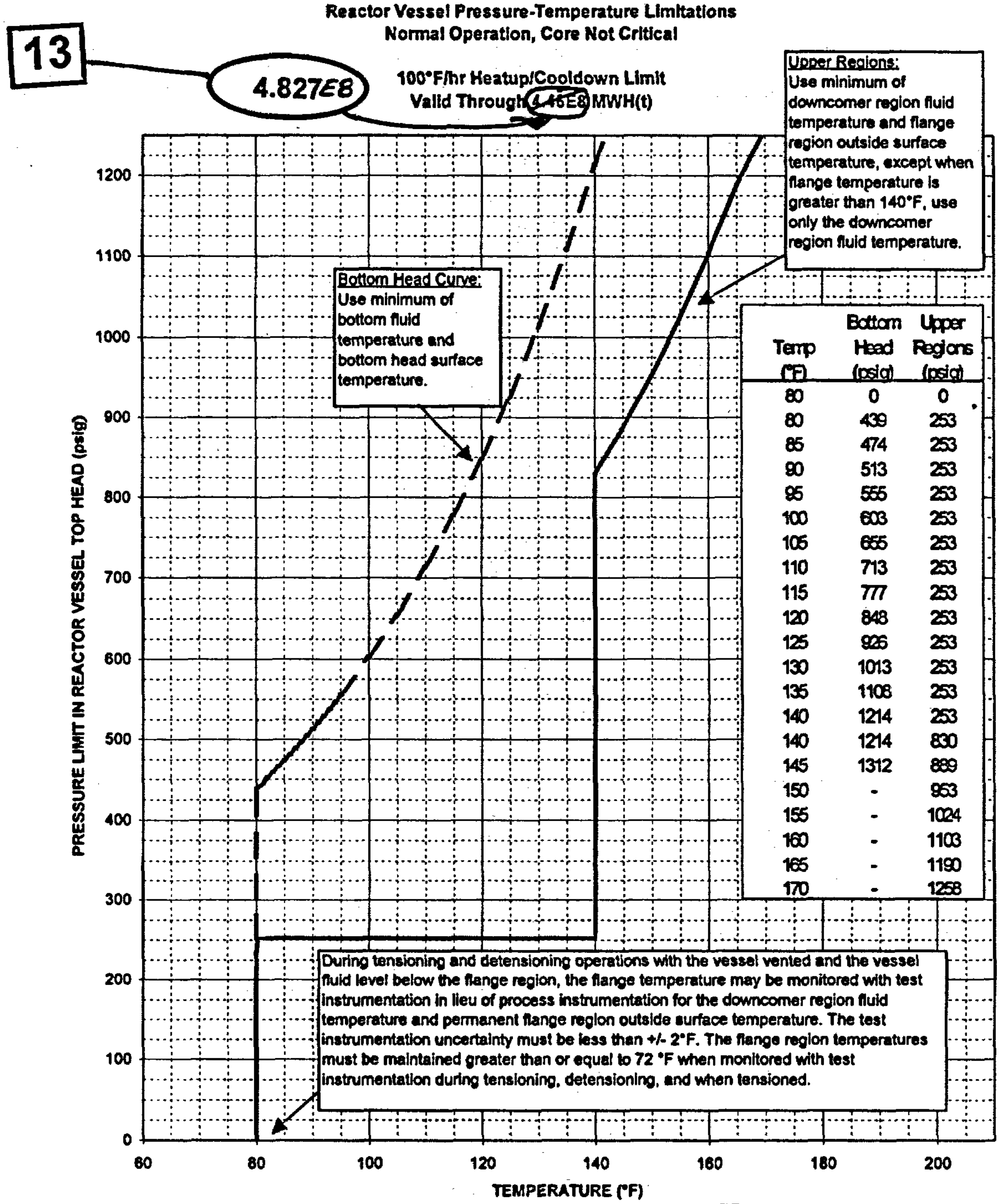
FIGURE 3.6.2

Reactor Vessel Pressure-Temperature Limitations
Normal Operation, Core Not Critical



Temp (°F)	Bottom Head (psig)	Upper Regions (psig)
80	0	0
80	439	253
85	474	253
90	513	253
95	555	253
100	603	253
105	655	253
110	713	253
115	777	253
120	848	253
125	926	253
130	1013	253
135	1108	253
140	1214	253
140	1214	830
145	1312	889
150	-	953
155	-	1024
160	-	1103
165	-	1190
170	-	1258

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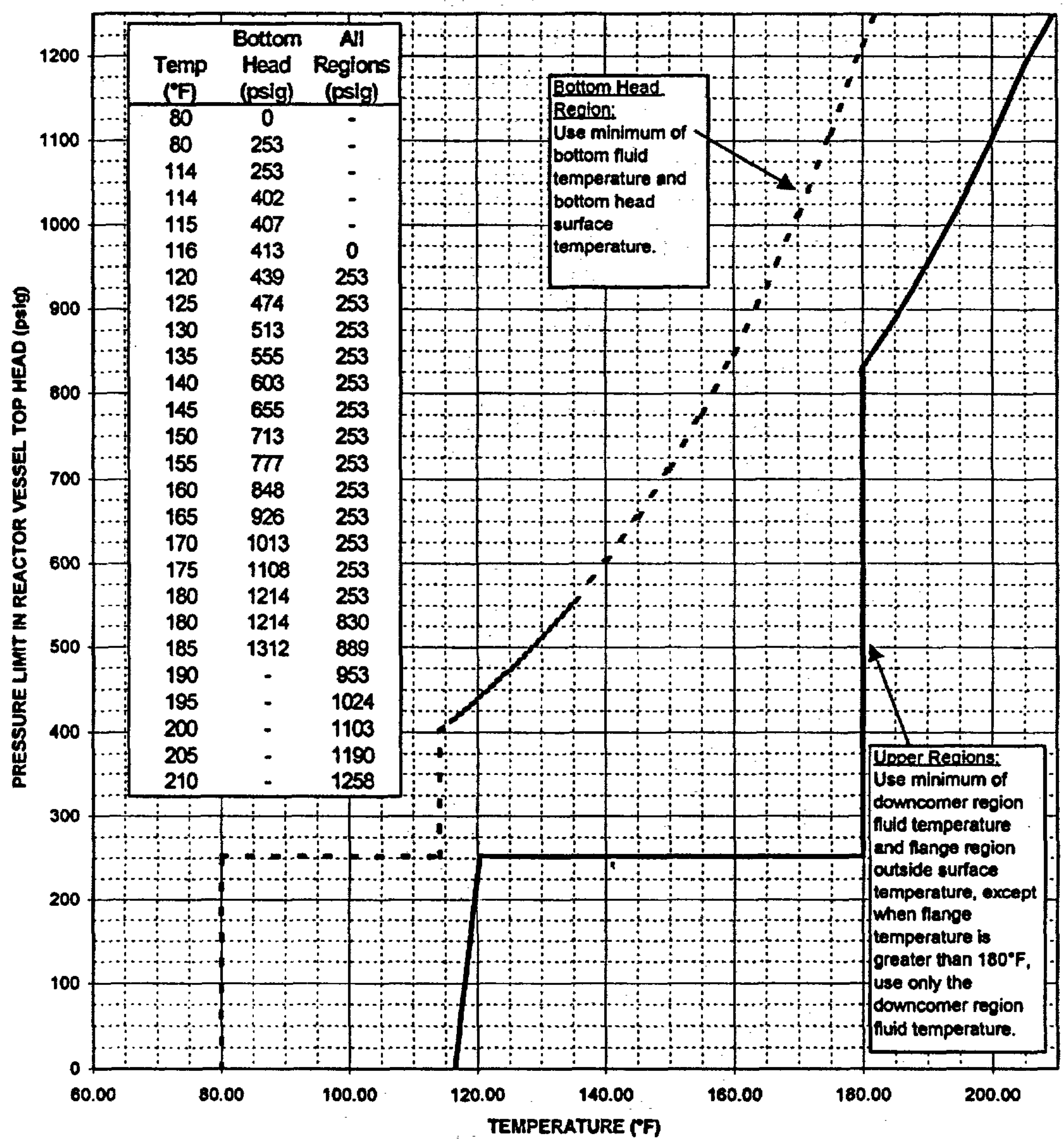
FIGURE 3.6.3

Reactor Vessel Pressure-Temperature Limitations
Normal Operation, Core Critical

100°F/hr Heatup/Cooldown Limit
If Pressure < 253 psig, Water Level must be within
Normal Range for Power Operation
Valid Through 4.46E8 MWH(t)

4.827E8

13



Temp (°F)	Bottom Head (psig)	All Regions (psig)
80	0	-
80	253	-
114	253	-
114	402	-
115	407	-
116	413	0
120	439	253
125	474	253
130	513	253
135	555	253
140	603	253
145	655	253
150	713	253
155	777	253
160	848	253
165	926	253
170	1013	253
175	1108	253
180	1214	253
180	1214	830
185	1312	889
190	-	953
195	-	1024
200	-	1103
205	-	1190
210	-	1258

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BASES:

3.6 and 4.6 REACTOR COOLANT SYSTEM

A. Pressure and Temperature Limitations

All components in th...
the effects o...
changes.
reactor
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Section 4...
temperature...
specified heat...
assumptions and sati...

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The Pressure/Temperature (P/T) curves included as Figures 3.6.1, 3.6.2, and 3.6.3 were developed using 10CFR50 Appendix G, 1995 ASME Code, Section XI, Appendix G (including the Summer 1996 Addenda), and ASME Code Case N-640. These three curves provide P/T limit requirements for Pressure Test, Core Not Critical, and Core Critical. The P/T curves are not derived from Design Basis Accident analysis. They are prescribed to avoid encountering pressure, temperature or temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the reactor pressure boundary, a condition that is unanalyzed.

During heating events, the thermal gradients in the reactor vessel wall produce thermal stresses that vary from compressive at the inner wall to tensile at the outer wall. During cooling events the thermal stresses vary from tensile at the inner wall to compressive at the outer wall. The thermally induced tensile stresses are additive to the pressure induced tensile stresses. In the flange region, bolt preload has a significant affect on stress in the flange and adjacent plates. Therefore heating/cooling events and bolt preload are used in the determination of the pressure-temperature limitations for the vessel.

The guidance of Branch Technical Position - MTEB 5-2, material drop weight, and Charpy impact test results were used to determine a reference nil-ductility temperature (RT_{NDT}) for all pressure boundary components. For the plates and welds adjacent to the core, fast neutron (E > 1 Mev) irradiation will cause an increase in the RT_{NDT}. For these plates and welds an adjusted RT_{NDT} (ART_{NDT}) of 89°F and 73°F (¼ and ¾ thickness locations) was conservatively used in development of these curves for core region components. Based upon plate and weld chemistry, initial RT_{NDT} values, predicted peak fast neutron fluence (~~2.99~~ 3.18 x 10^{17} n/cm^2 at the reactor vessel inside surface) for a gross power generation of ~~4.46~~ 4.827 x 10^8 MWH(t), these core region ART_{NDT} values conservatively bound the gui... Regulatory Guide 1.99, Revision 2.

20

There were five regions of ... pressure vessel (RPV) that were evaluated in the development of the P/T Limit curves: (1) the reactor vessel beltline region, (2) the bottom head region, (3) the feedwater nozzle, (4) the recirculation inlet nozzle, and (5) the upper vessel flange region. These regions will bound all other regions in the vessel with respect to considerations for brittle fracture.

Two lines are shown on each P/T limit figure. The dashed line is the Bottom Head Curve. This is applicable to the bottom head area only and includes the bottom head knuckle plates and dollar plates. Based on bottom head fluid temperature and bottom head surface temperature, the reactor pressure shall be maintained below the dashed line at all times.

BASES: 3.6 and 4.6 (Cont'd)

ensure compliance with the MCPR safety limit for the analyzed transients.

C. Coolant Leakage

The 5 gpm limit for unidentified leaks was established assuming such leakage was coming from the reactor coolant system. Tests have been conducted which demonstrate that a relationship exists between the size of a crack and the probability that the crack will propagate. These tests suggest that for leakage somewhat greater than the limit specified for unidentified leakage, the probability is small that imperfections or cracks associated with such leakage would grow rapidly. Leakage less than the limit specified can be detected within a few hours utilizing the available leakage detection systems. If the limit is exceeded and the origin cannot be determined in a reasonably short time the plant should be shutdown to allow further investigation and corrective action.

The 2 gpm increase limit in any 24 hour period for unidentified leaks was established as an additional requirement to the 5 gpm limit by Generic Letter 88-01, "NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping."

The removal capacity from the drywell floor drain sump and the equipment drain sump is 50 gpm each. Removal of 50 gpm from either of these sumps can be accomplished with considerable margin.

D. Safety and Relief Valves

Safety analyses have shown that only three of the four relief valves are required to ~~provide the recommended pressure margin of 25 psi below the safety valve actuation settings as well as compliance with the MCPR safety limit for the limiting anticipated overpressure transient.~~ For the purposes of this limiting condition, a relief valve that is unable to actuate within tolerance of its set pressure is considered to be as inoperable as a mechanically malfunctioning valve.

20

<MOVE>

The setpoint tolerance value for as-left or refurbished valves is specified in Section III of the ASME Boiler and Pressure Vessel Code as ±1% of set pressure. However, the code allows a larger tolerance value for the as-found condition if the supporting design analyses demonstrate that the applicable acceptance criteria are met. Safety analysis has been performed which shows that with all safety and safety relief valves within ±3% of the specified set pressures in Table 2.2.1 and with one inoperable safety relief valve, the reactor coolant pressure safety limit of 1375 psig and the MCPR safety limit are not exceeded during the limiting overpressure transient.

3.11 LIMITING CONDITIONS FOR OPERATION

3.11 REACTOR FUEL ASSEMBLIES

Applicability:

The Limiting Conditions for Operation associated with the fuel rods apply to these parameters which monitor the fuel rod operating conditions.

Objective:

The Objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.

Specifications:

A. Average Planar Linear Heat Generation Rate (APLHGR)

During operation at ~~≥25%~~ Rated Thermal Power, the APLHGR for each type of fuel as a function of average planar exposure, power, and flow shall not exceed the limiting values provided in the Core Operating Limits Report. For single recirculation loop operation, the limiting values shall be the values provided in the Core Operating Limits Report listed under the heading "Single Loop Operation." If at any time during operation at ~~≥25%~~ Rated Thermal Power it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, APLHGR(s) shall be returned to within prescribed limits within two (2) hours; otherwise, the reactor shall be brought to ~~<25%~~ Rated Thermal Power within 4 hours. Surveillance and corresponding action shal[l] until r[…] is wit[…] prescr[…]

14

23%

4.11 SURVEILLANCE REQUIREMENTS

4.11 REACTOR FUEL ASSEMBLIES

Applicability:

The Surveillance Requirements apply to the parameters which monitor the fuel rod operating conditions.

Objective:

The Objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.

Specifications:

A. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure, power, and flow shall be determined once within 12 hours after ~~≥25%~~ Rated Thermal Power and daily during operation at ~~≥25%~~ Rated Thermal Power thereafter.

23%

15

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3.11 LIMITING CONDITIONS FOR OPERATION

B. Linear Heat Generation Rate (LHGR)

16

23%

During operation at ≥25% Rated Thermal Power, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the maximum allowable LHGR provided in the Core Operating Limits Report.

If at any time during operation at ≥25% Rated Thermal Power it is determined by normal surveillance that the limiting value for LHGR is being exceeded, LHGR(s) shall be returned to within the prescribed limits within two (2) hours; otherwise, the reactor shall be brought to <25% Rated Thermal Power within 4 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

4.11 SURVEILLANCE REQUIREMENTS

B. Linear Heat Generation Rate (LHGR)

23%

17

The LHGR as a function of core height shall be checked once within 12 hours after ≥25% Rated Thermal Power and daily during operation at ≥25% Rated Thermal Power thereafter.

3.11 LIMITING CONDITIONS FOR OPERATION

C. Minimum Critical Power Ratio (MCPR)

18

23%

1. During operation at ≥25% Rated Thermal Power the MCPR operating value shall be equal to or greater than the MCPR limits provided in the Core Operating Limits Report. For single recirculation loop operation, the MCPR Limits at rated flow are also provided in the Core Operating Limits Report. If at any time during operation at ≥25% Rated Thermal Power it is determined by normal surveillance that the limiting value for MCPR is being exceeded, MCPR(s) shall be returned to within the prescribed limits within two (2) hours; otherwise, the reactor power shall be brought to <25% Rated Thermal Power within 4 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

4.11 SURVEILLANCE REQUIREMENTS

C. Minimum Critical Power Ratio (MCPR)

23%

19

MCPR shall be determined once within 12 hours after ≥25% Rated Thermal Power and daily during operation at ≥25% Rated Thermal Power thereafter.

NOTE: THIS PAGE IS NOT FROM CURRENT VY TECHNICAL SPECIFICATIONS, BUT IS AS PROPOSED IN VY LETTER OF MARCH 20, 2003, "IMPLEMENTATION OF ARTS/MELLLA AT VERMONT YANKEE," BVY 03-23.

BASES:

4.11 FUEL RODS

A. The APLHGR, LHGR and MCPR shall be checked daily when operating at ≥25% Rated Thermal Power to determine if fuel burnup, or control rod movement has caused changes in power distribution. Since changes due to burnup are slow, and only a few control rods are removed daily, a daily check of power distribution is adequate. For a limiting value to occur below 25% of rated thermal power, an unreasonably large peaking factor would be required, which is not the case for operating control rod sequences. The 12 hour allowance after thermal power ≥25% Rated Thermal Power is achieved is acceptable given the large inherent margin to operating limits at low power levels.

B. At certain times during plant startups and power changes the plant technical staff may determine that surveillance of APLHGR, LHGR and/or MCPR is necessary more frequently than daily. Because the necessity for such an augmented surveillance program is a function of a number of interrelated parameters, a reasonable program can only be determined on a case-by-case basis by the plant technical staff. The check of APLHGR, LHGR and MCPR will normally be done using the plant process computer. In the event that the computer is unavailable, the check will consist of either a manual calculation or a comparison of existing core conditions to those existing at the time of a previous check to determine if a significant change has occurred.

20

23%

If a reactor power distribution limit is exceeded, an assumption regarding an initial condition of the DBA analysis, transient analyses, or the fuel design analysis may not be met. Therefore, prompt action should be taken to restore the APLHGR, LHGR or MCPR to within the required limits such that the plant operates within analyzed conditions and within design limits of the fuel rods. The 2 hour completion time is sufficient to restore the APLHGR, LHGR, or MCPR to within its limits and is acceptable based on the low probability of a transient or DBA occurring simultaneously with the APLHGR, LHGR, or MCPR out of specification.

C. Minimum Critical Power Ratio (MCPR) - Surveillance Requirement

At core thermal power levels less than or equal to 25%, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. During initial start-up testing of the plant, a MCPR evaluation will be made at 25% thermal power level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR above 25% rated thermal power is sufficient since power distribution shifts are very slow during normal operation.

NOTE: THIS PAGE IS NOT FROM CURRENT VY TECHNICAL SPECIFICATIONS, BUT IS AS PROPOSED IN VY LETTER OF MARCH 20, 2003, "IMPLEMENTATION OF ARTS/MELLLA AT VERMONT YANKEE," BVY 03-23.

Docket No. 50-271
BVY 03-80

Attachment 10

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 263

Extended Power Uprate

Re-typed Technical Specification Pages

Listing of Affected Technical Specifications Pages

Replace the Vermont Yankee Nuclear Power Station Technical Specifications pages listed below with the revised pages included herein. The revised pages contain vertical lines in the margin indicating the areas of change. Also included is a change to page 3 of the Operating License to revise the authorized power level from 1593 megawatts thermal to 1912 megawatts thermal.

Current Page	New Page
OL 3	OL 3
3	3
6*	6*
7*	7*
10	10
11*	11*
12	12
14*	14*
15	15
17	17
21*	21*
24*	24*
30	30
83	83
90*	90*
92*	92*
94	94
97*	97*
98*	98*
135*	135*
136*	136*
137*	137*
138*	138*
142	142
224*	224*
225	225
226*	226*
228*	228*

* - Not current Technical Specifications—assumes acceptance of a prior license amendment request

E. Entergy Nuclear Operations, Inc., pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not to separate, such byproduct and special nuclear material as may be produced by operation of the facility.

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

   A. Maximum Power Level

   Entergy Nuclear Operations, Inc. is authorized to operate the facility at reactor core power levels not to exceed 1912 megawatts thermal in accordance with the Technical Specifications (Appendix A) appended hereto.

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment 208, are hereby incorporated in the license. Entergy Nuclear Operations, Inc. shall operate the facility in accordance with the Technical Specifications.

   C. Reports

   Entergy Nuclear Operations, Inc. shall make reports in accordance with the requirements of the Technical Specifications.

   D. Records

   Entergy Nuclear Operations, Inc. shall keep facility operating records in accordance with the requirements of the Technical Specifications.

   E. Environmental Conditions

   Pursuant to the Initial Decision of the presiding Atomic Safety and Licensing Board issued February 27, 1973, the following conditions for the protection of the environment are incorporated herein:

or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system or the coincident tripping of two trip systems.

3. Protective Action - An action initiated by the protection system when a limit is reached. A protective action can be at a channel or system level.

4. Protective Function - A system protective action which results from the protective action of the channels monitoring a particular plant condition.

P. Rated Neutron Flux - Rated neutron flux is the neutron flux that corresponds to a steady state power level of 1912 thermal megawatts.

Q. Rated Thermal Power - Rated thermal power means a steady state power level of 1912 thermal megawatts.

R. Reactor Power Operation - Reactor power operation is any operation with the mode switch in the "Startup/Hot Standby" or "Run" position with the reactor critical and above 1% rated thermal power.

1. Startup/Hot Standby Mode - In this mode the low turbine condenser vacuum trip is bypassed when condenser vacuum is less than 12 inches Hg and both turbine stop valves and bypass valves are closed; the low pressure and the 10 percent closure main steamline isolation valve closure trips are bypassed; the reactor protection system is energized with IRM neutron monitoring system trips and control rod withdrawal interlocks in service and APRM neutron monitoring system operable.

2. Run Mode - In this mode the reactor system pressure is equal to or greater than 800 psig and the reactor protection system is energized with APRM protection and RBM interlocks in service.

S. Reactor Vessel Pressure - Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space detector.

T. Refueling Outage - Refueling outage is the period of time between the shutdown of the unit prior to a refueling and the startup of the plant subsequent to that refueling. For the purpose of designating frequency of testing and surveillance, a refueling outage shall mean a regularly scheduled refueling outage; however, where such outages occur within 8 months of the completion of the previous refueling outage, the required surveillance testing need not be performed until the next regularly scheduled outage.

U. Deleted

1.1 SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

Applicability:

Applies to the interrelated variable associated with fuel thermal behavior.

Objective:

To establish limits below which the integrity of the fuel cladding is preserved.

Specification:

A. Bundle Safety Limit (Reactor Pressure >800 psia and Core Flow >10% of Rated)

When the reactor pressure is >800 psia and the core flow is greater than 10% of rated:

1. A Minimum Critical Power Ratio (MCPR) of less than 1.10 (1.12 for Single Loop Operation) shall constitute violation of the Fuel Cladding Integrity Safety Limit (FCISL).

2.1 LIMITING SAFETY SYSTEM SETTING

2.1 FUEL CLADDING INTEGRITY

Applicability:

Applies to trip setting of the instruments and devices which are provided to prevent the nuclear system safety limits from being exceeded.

Objective:

To define the level of the process variable at which automatic protective action is initiated.

Specification:

A. Trip Settings

The limiting safety system trip settings shall be as specified below:

1. Neutron Flux Trip Settings

a. APRM Flux Scram Trip Setting (Run Mode)

When the mode switch is in the RUN position, the APRM flux scram trip setting shall be as shown on Figure 2.1.1 and shall be:

Two loop operation:

$S \leq 0.33W + 53.7\%$ for $0\% < W \leq 30.9\%$

$S \leq 1.07W + 30.8\%$ for $30.9\% < W \leq 66.7\%$

$S \leq 0.55W + 65.5\%$ for $66.7\% < W \leq 99.0\%$

With a maximum of 120.0% power for $W > 99.0\%$

Single loop operation:

$S \leq 0.33W + 51.1\%$ for $0\% < W \leq 39.1\%$

$S \leq 1.07W + 22.2\%$ for $39.1\% < W \leq 61.7\%$

$S \leq 0.55W + 54.3\%$ for $61.7\% < W \leq 119.4\%$

With a maximum of 120.0% power for $W > 119.4\%$

where:

S = setting in percent of rated thermal power (1912 MWt)

1.1 SAFETY LIMIT

B. Core Thermal Power Limit (Reactor Pressure $\leq$ 800 psia or Core Flow $\leq$ 10% of Rated)

When the reactor pressure is $\leq$800 psia or core flow $\leq$10% of rated, the core thermal power shall not exceed 23% of rated thermal power.

C. Power Transient

To ensure that the safety limit established in Specification 1.1A and 1.1B is not exceeded, each required scram shall be initiated by its expected scram signal. The safety limit shall be assumed to be exceeded when scram is accomplished by means other than the expected scram signal.

D. Whenever the reactor is shutdown with irradiated fuel in the reactor vessel, the water level shall not be less than 12 inches above the top of the enriched fuel when it is seated in the core.

2.1 LIMITING SAFETY SYSTEM SETTING

W = percent rated two loop drive flow where 100% rated drive flow is that flow equivalent to 48 x 10^6 lbs/hr core flow

In the event of operation at >23% Rated Thermal Power the APRM gain shall be equal to or greater than 1.0.

1.1 SAFETY LIMIT

2.1 LIMITING SAFETY SYSTEM SETTING

D. Reactor low-low water level Emergency Core Cooling System (ECCS) initiation shall be at least 82.5 inches above the top of the enriched fuel.

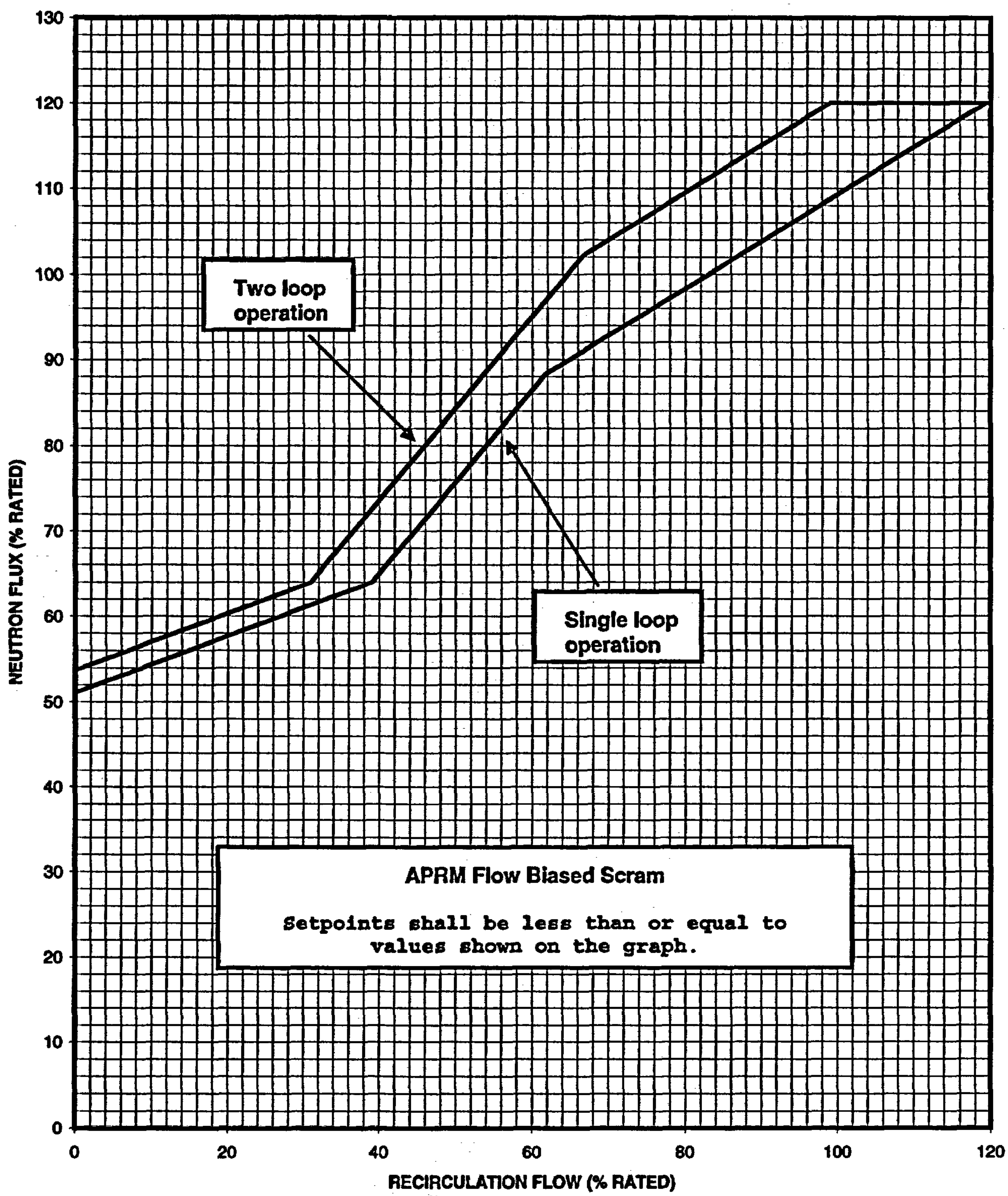
E. Turbine stop valve scram shall, when operating at greater than 25% of Rated Thermal Power, be less than or equal to 10% valve closure from full open.

F. Turbine control valve fast closure scram shall, when operating at greater than 25% of Rated Thermal Power, trip upon actuation of the turbine control valve fast closure relay.

G. Main steam line isolation valve closure scram shall be less than or equal to 10% valve closure from full open.

H. Main steam line low pressure initiation of main steam line isolation valve closure shall be at least 800 psig.

FIGURE 2.1.1
APRM FLOW REFERENCE SCRAM SETTING



BASES:

1.1 FUEL CLADDING INTEGRITY

A. Refer to General Electric Company Licensing Topical Report, "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A (most recent revision).

The fuel cladding integrity Safety Limit (SL) is set such that no significant fuel damage is calculated to occur if the limit is not violated. Since the parameters that result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions that result in the onset of transition boiling have been used to mark the beginning of the region in which fuel damage could occur. Although it is recognized that the onset of transition boiling would not result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity SL is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition, considering the power distribution within the core and all uncertainties.

The MCPR SL is determined using a statistical model that combines all the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved General Electric Critical Power correlations.

The MCPR fuel cladding integrity SL is increased for single loop operation in order to account for increased core flow measurement and TIP reading uncertainties.

B. Core Thermal Power Limit (Reactor Pressure ≤ 800 psia or Core Flow ≤10% of Rated)

At pressures below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low power and all flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and all flows will always be greater than 4.56 psi. Analyses show that with a flow of 28 x 10^3 lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than 28 x 10^3 lbs/hr irrespective of total core flow and independent of bundle power for the range of bundle powers of concern. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors this corresponds to a core thermal power of more than 50%. Thus, a core thermal power limit of 23% for reactor pressures below 800 psia or core flow less than 10% is conservative.

C. Power Transient

Plant safety analyses have shown that the scrams caused by exceeding any safety setting will assure that the Safety Limit of Specification 1.1.1A or 1.1.1B will not be exceeded. Scram times are checked periodically to assure the insertion times are adequate. The thermal power transient resulting when a scram is accomplished other than by the expected scram signal (e.g., scram from neutron flux

BASES:

2.1 FUEL CLADDING INTEGRITY

A. Trip Settings

The bases for individual trip settings are discussed in the following paragraphs.

1. Neutron Flux Trip Settings

a. APRM Flux Scram Trip Setting (Run Mode)

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady state conditions, reads in percent of rated thermal power (1912 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses are performed to demonstrate that the APRM flux scram over the range of settings from a maximum of 120% to the minimum flow biased setting provide protection from the fuel safety limit for all abnormal operational transients including those that may result in a thermal hydraulic instability.

An increase in the APRM scram trip setting would decrease the margin present before the fuel cladding integrity Safety Limit is reached. The APRM scram trip setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM scram trip setting was selected because it provides adequate margin for the fuel cladding integrity Safety Limit yet allows operating margin that reduces the possibility of unnecessary scrams. The relationship between recirculation drive flow and reactor core flow is non-linear at low core flows. Therefore, separate APRM flow biased scram trip setting equations are provided for low core flows.

The scram trip is set to ensure acceptable transient response. For single recirculation loop operation, the APRM flux scram trip setting is reduced in accordance with the analysis presented in NEDO-30060, February 1983. This adjustment accounts for the difference between the single loop and two loop drive flow at the same core flow, and ensures that the margin of safety is not reduced during single loop operation. The single loop operation equations are based on a bounding (maximum) difference between two loop and single loop drive flow at the same core flow of 8%.

Analyses of the limiting transients show that no scram adjustment is required to assure fuel cladding integrity when the transient is initiated from the operating limit MCPR defined in the Core Operating Limits Report.

<u>BASES</u>: 2.1 (Cont'd)

<u>Flux Scram Trip Setting (Refuel or Startup and Hot Standby Mode)</u>

For operation in the startup mode while the reactor is at low pressure, the reduced APRM scram setting to 15% of rated power provides adequate thermal margin between the setpoint and the safety limit, 23% of the rated. (During an outage when it is necessary to check refuel interlocks, the mode switch must be moved to the startup position. Since the APRM reduced scram may be inoperable at that time due to the disconnection of the LPRMs, it is required that the IRM scram and the SRM scram in noncoincidence be in effect. This will ensure that adequate thermal margin is maintained between the setpoint and the safety limit.) The margin is adequate to accommodate anticipated maneuvers associated with station startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5% of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The reduced APRM scram remains active until the mode switch is placed in the RUN position. This switch can occur when reactor pressure is greater than 800 psia.

The IRM system consists of 6 chambers, 3 in each of the reactor protection system logic channels. The IRM is a 5-decade instrument, which covers the range of power level between that covered by the SRM and the APRM. The 5 decades are covered by the IRM by means of a range switch and the 5 decades are broken down into 10 ranges, each being one-half of a decade in size. The IRM scram trip setting of 120/125 of full scale is active in each range of the IRM. For example, if the instrument were on range 1, the scram setting would be a 120/125 of full scale for that range; likewise, if the instrument were on range 5, the scram would be 120/125 of full scale on that range. Thus, as the IRM is ranged up to accommodate the increase in power level, the scram trip setting is also ranged up. The most significant sources of reactivity change during the power increase are due to control rod withdrawal. For in-sequence control rod withdrawal, the rate of change of power is slow enough due to the physical limitation of withdrawing control rods, that heat flux is in equilibrium with the neutron flux and an IRM scram would result in a reactor shutdown well before any safety limit is exceeded.

BASES: 2.1 (Cont'd)

E. Turbine Stop Valve Closure Scram Trip Setting

The turbine stop valve closure scram trip anticipates the pressure, neutron flux and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of <10% of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above the fuel cladding integrity safety limit even during the worst case transient that assumes the turbine bypass is closed. This scram signal may be bypassed at ≤25% of reactor Rated Thermal Power.

F. Turbine Control Valve Fast Closure Scram

The control valve fast closure scram is provided to limit the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves due to a load rejection coincident with failure of the bypass system. This transient is less severe than the turbine stop valve closure with failure of the bypass valves and therefore adequate margin exists. This scram signal may be bypassed at ≤25% of reactor Rated Thermal Power.

G. Main Steam Line Isolation Valve Closure Scram

The isolation scram anticipates the pressure and flux transients which occur during an isolation event and the loss of inventory during a pipe break. This action minimizes the effect of this event on the fuel and pressure vessel.

H. Reactor Coolant Low Pressure Initiation of Main Steam Isolation Valve Closure

The low pressure isolation of the main steam lines at 800 psig is provided to give protection against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage is taken of the scram feature which occurs when the main steam line isolation valves are closed, to provide the reactor shutdown so that high power operation at low reactor pressure does not occur. Operation of the reactor at pressures lower than 800 psig requires that the reactor mode switch be in the startup position where protection of the fuel cladding integrity safety limit is provided by the IRM high neutron flux scram.

Thus, the combination of main steam line low pressure isolation and isolation valve closure scram assures the availability of neutron scram protection over the entire range of applicability of the fuel cladding integrity safety limit.

TABLE 3.1.1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT REQUIREMENTS

	Trip Function	Trip Settings	Modes in Which Functions Must be Operating: Refuel (1)	Startup (12)	Run	Minimum Number Operating Instrument Channels Per Trip System (2)	Required ACTIONS When Minimum Conditions For Operation Are Not Satisfied (3)
1.	Mode Switch in Shutdown (5A-S1)		X	X	X	1	A
2.	Manual Scram (5A-S3A/B)		X	X	X	1	A
3.	IRM (7-41(A-F))						
	High Flux	$\leq 120/125$	X	X		2	A
	INOP		X	X		2	A
4.	APRM (APRM A-F)						
	High Flux (flow bias)	Two loop operation: (4) $S \leq 0.33W + 53.7\%$ for $0\% < W \leq 30.9\%$ $S \leq 1.07W + 30.8\%$ for $30.9\% < W \leq 66.7\%$ $S \leq 0.55W + 65.5\%$ for $66.7\% < W \leq 99.0\%$ With a maximum of 120.0% power for $W > 99.0\%$ Single loop operation: (4) $S \leq 0.33W + 51.1\%$ for $0\% < W \leq 39.1\%$ $S \leq 1.07W + 22.2\%$ for $39.1\% < W \leq 61.7\%$ $S \leq 0.55W + 54.3\%$ for $61.7\% < W \leq 119.4\%$ With a maximum of 120.0% power for $W > 119.4\%$			X	2	A or B
	High Flux (reduced)	$\leq 15\%$	X	X		2	
	INOP			X	X	2(5)	A A or B
5.	High Reactor Pressure (PT-2-3-55(A-D) (M)	≤ 1055 psig	X	X	X	2	A

TABLE 3.1.1 NOTES (Cont'd)

3. When the requirements in the column "Minimum Number of Operating Instrument Channels Per Trip System" cannot be met for one system, that system shall be tripped. If the requirements cannot be met for both trip systems, the appropriate ACTIONS listed below shall be taken:

   a) Initiate insertion of operable rods and complete insertion of all operable rods within four hours.
   b) Reduce power level to IRM range and place mode switch in the "Startup/Hot Standby" position within eight hours.
   c) Reduce turbine load and close main steam line isolation valves within 8 hours.
   d) Reduce reactor power to less than 25% of rated within 8 hours.

4. "W" is percent rated two loop drive flow where 100% rated drive flow is that flow equivalent to 48 x 10^6 lbs/hr core flow.

5. To be considered operable an APRM must have at least 2 LPRM inputs per level and at least a total of 13 LPRM inputs, except that channels A, C, D, and F may lose all LPRM inputs from the companion APRM Cabinet plus one additional LPRM input and still be considered operable.

6. The top of the enriched fuel has been designated as 0 inches and provides common reference level for all vessel water level instrumentation.

7. Deleted.

8. Deleted.

9. Channel signals for the turbine control valve fast closure trip shall be derived from the same event or events which cause the control valve fast closure.

10. Turbine stop valve closure and turbine control valve fast closure scram signals may be bypassed at $\leq$25% of reactor Rated Thermal Power.

11. Not used.

12. While performing refuel interlock checks which require the mode switch to be in Startup, the reduced APRM high flux scram need not be operable provided:

    a. The following trip functions are operable:

       1. Mode switch in shutdown,
       2. Manual scram,
       3. High flux IRM scram
       4. High flux SRM scram in noncoincidence,
       5. Scram discharge volume high water level, and;

    b. No more than two (2) control rods withdrawn. The two (2) control rods that can be withdrawn cannot be face adjacent or diagonally adjacent.

BASES: 3.1 (Cont'd)

Instrumentation is provided to detect a loss-of-coolant accident and initiate the core standby cooling equipment. This instrumentation is a backup to the water level instrumentation which is discussed in Specification 3.2.

The Control Rod Drive Scram System is designed so that all of the water that is discharged from the reactor by the scram can be accommodated in the discharge piping. This discharge piping is divided into two sections. One section services the control rod drives on the north side of the reactor, the other serves the control rod drives of the south side. A part of the piping in each section is an instrument volume which accommodates in excess of 21 gallons of water and is at the low point in the piping. No credit was taken for this volume in the design of the discharge piping as concerns the amount of water which must be accommodated during a scram. During normal operation, the discharge volume is empty; however, should it fill with water, the water discharged to the piping from the reactor could not be accommodated, which would result in slow scram times or partial or no control rod insertion. To preclude this occurrence, level instrumentation has been provided for the instrument volume which scram the reactor when the volume of water reaches 21 gallons. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharged water, and precludes the situation in which a scram would be required but not be able to perform its function adequately. The present design of the Scram Discharge System is in concert with the BWR Owner's Group criteria, which have previously been endorsed by the NRC in their generic "Safety Evaluation Report (SER) for Scram Discharge Systems", dated December 1, 1980.

Loss of condenser vacuum occurs when the condenser can no longer handle the heat input. Loss of condenser vacuum initiates a closure of the turbine stop valves and turbine bypass valves which eliminates the heat input to the condenser. Closure of the turbine stop and bypass valves causes a pressure transient, neutron flux rise, and an increase in surface heat flux. To prevent the clad safety limit from being exceeded if this occurs, a reactor scram occurs on turbine stop valve closure. The turbine stop valve closure scram function alone is adequate to prevent the clad safety limit from being exceeded in the event of a turbine trip transient without bypass.

Turbine stop valve (TSV) closure and turbine control valve (TCV) fast closure scram signals may be bypassed at ≤25% of reactor Rated Thermal Power since, at low thermal power levels, the margins to fuel thermal-hydraulic limits and reactor primary coolant boundary pressure limits are large and an immediate scram is not necessary. This bypass function is normally accomplished automatically by pressure switches sensing turbine first stage pressure. The turbine first stage pressure setpoint controlling the bypass of the scram signals on TCV fast closure and TSV closure is derived from analysis of reactor pressurization transients. Certain operational factors, such as turbine bypass valves open, can influence the relationship between turbine first stage pressure and reactor Rated Thermal Power. However, above 25% of reactor Rated Thermal Power, these scram functions must be enabled.

3.3 LIMITING CONDITIONS FOR OPERATION

2. The Control Rod Drive Housing Support System shall be in place when the Reactor Coolant System is pressurized above atmospheric pressure with fuel in the reactor vessel unless all operable control rods are fully inserted.

3. While the reactor is below 17% power, the Rod Worth Minimizer (RWM) shall be operating while moving control rods except that:

   (a) If after withdrawal of at least 12 control rods during a startup, the RWM fails, the startup may continue provided a second licensed operator verifies that the operator at the reactor console is following the control rod program; or

   (b) If all rods, except those that cannot be moved with control rod drive

4.3 SURVEILLANCE REQUIREMENTS

positive coupling and the results of each test shall be recorded. The drive and blade shall be coupled and fully withdrawn. The position and over-travel lights shall be observed.

2. The Control Rod Drive Housing Support System shall be inspected after reassembly and the results of the inspection recorded.

3. Prior to control rod withdrawal for startup the Rod Worth Minimizer (RWM) shall be verified as operable by performing the following:

   (a) Verify that the control rod withdrawal sequence for the Rod Worth Minimizer computer is correct.

   (b) The Rod Worth Minimizer diagnostic test shall be performed.

BASES: 3.3 & 4.3 (Cont'd)

2. The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage of the primary coolant system. The design basis is given in Subsection 3.5.2 of the FSAR, and the design evaluation is given in Subsection 3.5.4. This support is not required if the reactor coolant system is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing.

3. In the course of performing normal startup and shutdown procedures, a pre-specified sequence for the withdrawal or insertion of control rods is followed. Control rod dropout accidents which might lead to significant core damage, cannot occur if this sequence of rod withdrawals or insertions is followed. The Rod Worth Minimizer restricts withdrawals and insertions to those listed in the pre-specified sequence and provides an additional check that the reactor operator is following prescribed sequence. Although beginning a reactor startup without having the RWM operable would entail unnecessary risk, continuing to withdraw rods if the RWM fails subsequently is acceptable if a second licensed operator verifies the withdrawal sequence. Continuing the startup increases core power, reduces the rod worth and reduces the consequences of dropping any rod. Withdrawal of rods for testing is permitted with the RWM inoperable, if the reactor is subcritical and all other rods are fully inserted. Above 17% power, the RWM is not needed since even with a single error an operator cannot withdraw a rod with sufficient worth, which if dropped, would result in anything but minor consequences.

4. Refer to the "General Electric Standard Application for Reactor Fuel (GESTAR II)," NEDE-24011-P-A, (the latest NRC-approved version will be listed in the COLR).

5. The Source Range Monitor (SRM) system provides a scram function in noncoincident configuration. It does provide the operator with a visual indication of neutron level. The consequences of reactivity accidents are a function of the initial neutron flux. The requirement of at least three counts per second assures that any transient, should it occur, begins at or above the initial value of 10^{-8} of rated power used in the analyses of transients from cold conditions. One operable SRM channel is adequate to monitor the approach to criticality, therefore, two operable SRM's are specified for added conservatism.

6. Deleted.

3.4 LIMITING CONDITIONS FOR OPERATION

3.4 REACTOR STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the operating status of the Reactor Standby Liquid Control System.

Objective:

To assure the availability of an independent reactivity control mechanism.

Specification:

A. Normal Operation

Except as specified in 3.4.B below, the Standby Liquid Control System shall be operable when the reactor mode switch is in either the "Startup/Hot Standby" or "Run" position, except to allow testing of instrumentation associated with the reactor mode switch interlock functions provided:

1. Reactor coolant temperature is less than or equal to 212° F;
2. All control rods remain fully inserted in core cells containing one or more fuel assemblies; and
3. No core alterations are in progress.

4.4 SURVEILLANCE REQUIREMENTS

4.4 REACTOR STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the periodic testing requirement for the Reactor Standby Liquid Control System.

Objective:

To verify the operability of the Standby Liquid Control System.

Specification:

A. Normal Operation

The Standby Liquid Control System shall be verified operable by:

1. Testing pumps and valves in accordance with Specification 4.6.E. A minimum flow rate of 35 gpm at ≥ 1325 psig shall be verified for each pump.
2. Verifying the continuity of the explosive charges at least monthly.

In addition, at least once during each operating cycle, the Standby Liquid Control System shall be verified operable by:

3. Testing that the setting of the pressure relief valves is between 1400 and 1490 psig.
4. Initiating one of the standby liquid control loops, excluding the primer chamber and inlet fitting, and verifying that a flow path from a pump to the reactor vessel is available. Both loops shall be tested over the course of two operating cycles.

3.4 LIMITING CONDITIONS FOR OPERATION

2. The solution temperature, including that in the pump suction piping, shall be maintained above the curve shown in Figure 3.4.2.

3. The combination of Standby Liquid Control System pump flow rate, boron concentration, and boron enrichment shall satisfy the following relationship for the Standby Liquid Control System to be considered operable:

$$\frac{Q}{86} \times \frac{M251}{M} \times \frac{C}{13} \times \frac{E}{19.8} \geq 1.29$$

where:

C = the concentration of sodium pentaborate solution (weight percent) in the Standby Liquid Control System tank

E = the boron-10 enrichment (atom percent) of the sodium pentaborate solution

Q = ≥ 35 gpm

$\frac{M251}{M}$ = a constant (the ratio of mass of water in the reference plant compared to VY)

D. If Specification 3.4.A or B is not met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

E. If Specification 3.4.C is not met, action shall be immediately initiated to correct the deficiency. If at the end of 12 hours the system has not been restored to full operability, then a shutdown shall be initiated with the reactor in cold shutdown within 24 hours of initial discovery.

4.4 SURVEILLANCE REQUIREMENTS

2. Sodium pentaborate concentration shall be determined at least once a month and within 24 hours following the addition of water or boron, or if the solution temperature drops below the limits specified by Figure 3.4.2.

3. The boron-10 enrichment of the borated solution required by Specification 3.4.C.3 shall be tested and verified once per operating cycle.

BASES:

3.4 & 4.4 REACTOR STANDBY LIQUID CONTROL SYSTEM

A. Normal Operation

The design objective of the Reactor Standby Liquid Control System (SLCS) is to provide the capability of bringing the reactor from full power to a cold, xenon-free shutdown assuming that none of the withdrawn control rods can be inserted. To meet this objective, the Standby Liquid Control System is designed to inject a quantity of boron which produces a concentration of 800 ppm of natural boron in the reactor core in less than 138 minutes. An 800 ppm natural boron concentration in the reactor core is required to bring the reactor from full power to a 5% Δk subcritical condition. An additional margin (25% of boron) is added for possible imperfect mixing of the chemical solution in the reactor water. A minimum quantity of 3850 gallons of solution having a 10.1% natural sodium pentaborate concentration is required to meet this shutdown requirement.

The time requirement (138 minutes) for insertion of the boron solution was selected to override the rate of reactivity insertion due to cooldown of the reactor following the xenon poison peak. For a required minimum pumping rate of 35 gallons per minute, the maximum net storage volume of the boron solution is established as 4830 gallons.

In addition to its original design basis, the Standby Liquid Control System also satisfies the requirements of 10CFR50.62(c)(4) on anticipated transients without scram (ATWS) by using enriched boron. The ATWS rule adds hot shutdown and neutron absorber (i.e., boron-10) injection rate requirements that exceed the original Standby Liquid Control System design basis. However, changes to the Standby Liquid Control System as a result of the ATWS rule have not invalidated the original design basis.

With the reactor mode switch in the "Run" or "Startup/Hot Standby" position, shutdown capability is required. With the mode switch in "Shutdown," control rods are not able to be withdrawn since a control rod block is applied. This provides adequate controls to ensure that the reactor remains subcritical. With the mode switch in "Refuel," only a single control rod can be withdrawn from a core cell containing fuel assemblies. Determination of adequate shutdown margin by Specification 3.3.A ensures that the reactor will not become critical. Therefore, the Standby Liquid Control System is not required to be operable when only a single control rod can be withdrawn.

Pump operability testing (by recirculating demineralized water to the test tank)in accordance with Specification 4.6.E is adequate to detect if failures have occurred. Flow, relief valve, circuitry, and trigger assembly testing at the prescribed intervals assures a high reliability of system operation capability. The maximum SLCS pump discharge pressure during the limiting ATWS event is 1325 psig. This value is based on a reactor vessel lower plenum pressure of 1292 psia that occurs during the limiting ATWS event at the time of SLCS initiation, i.e., 120 seconds into the event. There is adequate margin to prevent the SLCS relief valve from lifting. With a nominal SLCS relief valve setpoint of 1400 psig, there is a margin of 75 psi between the peak SLCS pump discharge pressure and the relief valve nominal setpoint. Recirculation of the borated solution is done during each operating cycle to ensure one suction line from the boron tank is clear. In addition, at least once during each operating cycle, one of the standby liquid control loops will be initiated to verify that a flow path from a pump to the reactor vessel is available by pumping demineralized water into the reactor vessel.

BASES: 3.4 & 4.4 (Cont'd)

B. Operation With Inoperable Components

Only one of the two standby liquid control pumping circuits is needed for proper operation of the system. If one pumping circuit is found to be inoperable, there is no immediate threat to shutdown capability, and reactor operation may continue while repairs are being made. Assurance that the system will perform its intended function is obtained from the results of the pump and valve testing performed in accordance with ASME Section XI requirements

C. Standby Liquid Control System Tank - Borated Solution

The solution saturation temperature varies with the concentration of sodium pentaborate. The solution shall be kept at least 10°F above the saturation temperature to guard against boron precipitation. The 10°F margin is included in Figure 3.4.2. Temperature and liquid level alarms for the system are annunciated in the Control Room.

Once the solution has been made up, boron concentration will not vary unless more boron or water is added. Level indication and alarm indicate whether the solution volume has changed which might indicate a possible solution concentration change. Considering these factors, the test interval has been established.

Sodium pentaborate concentration is determined within 24 hours following the addition of water or boron, or if the solution temperature drops below specified limits. The 24-hour limit allows for 8 hours of mixing, subsequent testing, and notification of shift personnel.

Boron concentration, solution temperature, and volume are checked on a frequency to assure a high reliability of operation of the system should it ever be required. Isotopic tests of the sodium pentaborate are performed periodically to ensure that the proper boron-10 atom percentage is being used.

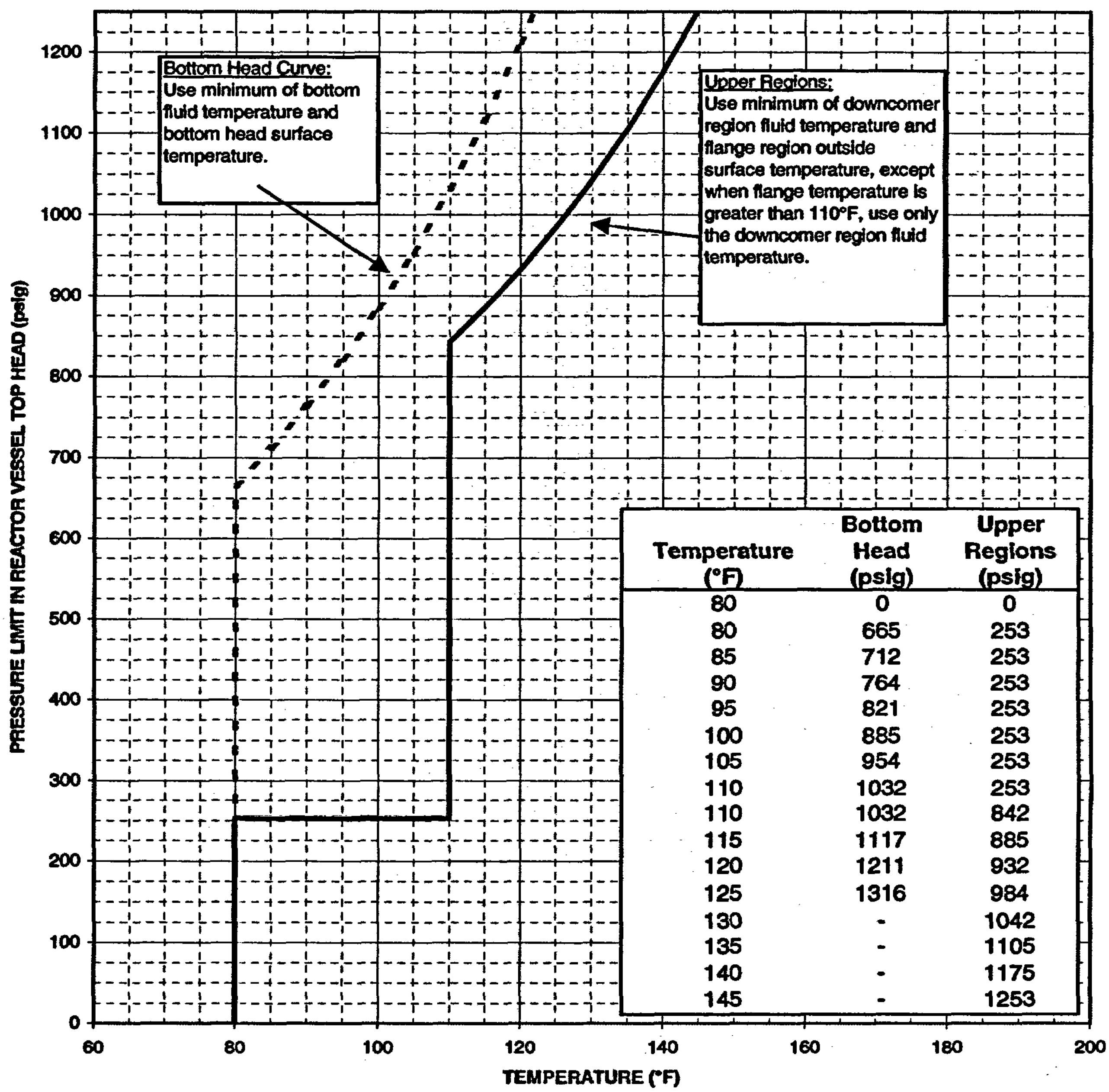
10CFR50.62(c)(4) requires a Standby Liquid Control System with a minimum flow capacity and boron content equivalent to 86 gpm of 13 weight percent natural sodium pentaborate solution in the 251-inch reactor pressure vessel reference plant. Natural sodium pentaborate solution is 19.8 atom percent boron-10. The relationship expressed in Specification 3.4.C.3 also contains the ratio M251/M to account for the difference in water volume between the reference plant and Vermont Yankee. (This ratio of masses is 628,300 lbs./401,247 lbs.)

To comply with the ATWS rule and the plant-specific ATWS analysis, the combination of three Standby Liquid Control System parameters must be considered: boron concentration, Standby Liquid Control System pump flow rate, and boron-10 enrichment. If the product of the expression in Specification 3.4.C.3 is equal to or greater than 1.29, the Standby Liquid Control System satisfies the requirements of 10CFR50.62(c)(4) and the plant-specific ATWS analysis.

Figure 3.6.1

Reactor Vessel Pressure-Temperature Limitations
Hydrostatic Pressure and Leak Tests, Core Not Critical

40°F/hr Heatup/Cooldown Limit
Valid Through 4.627E6 MWH(t)

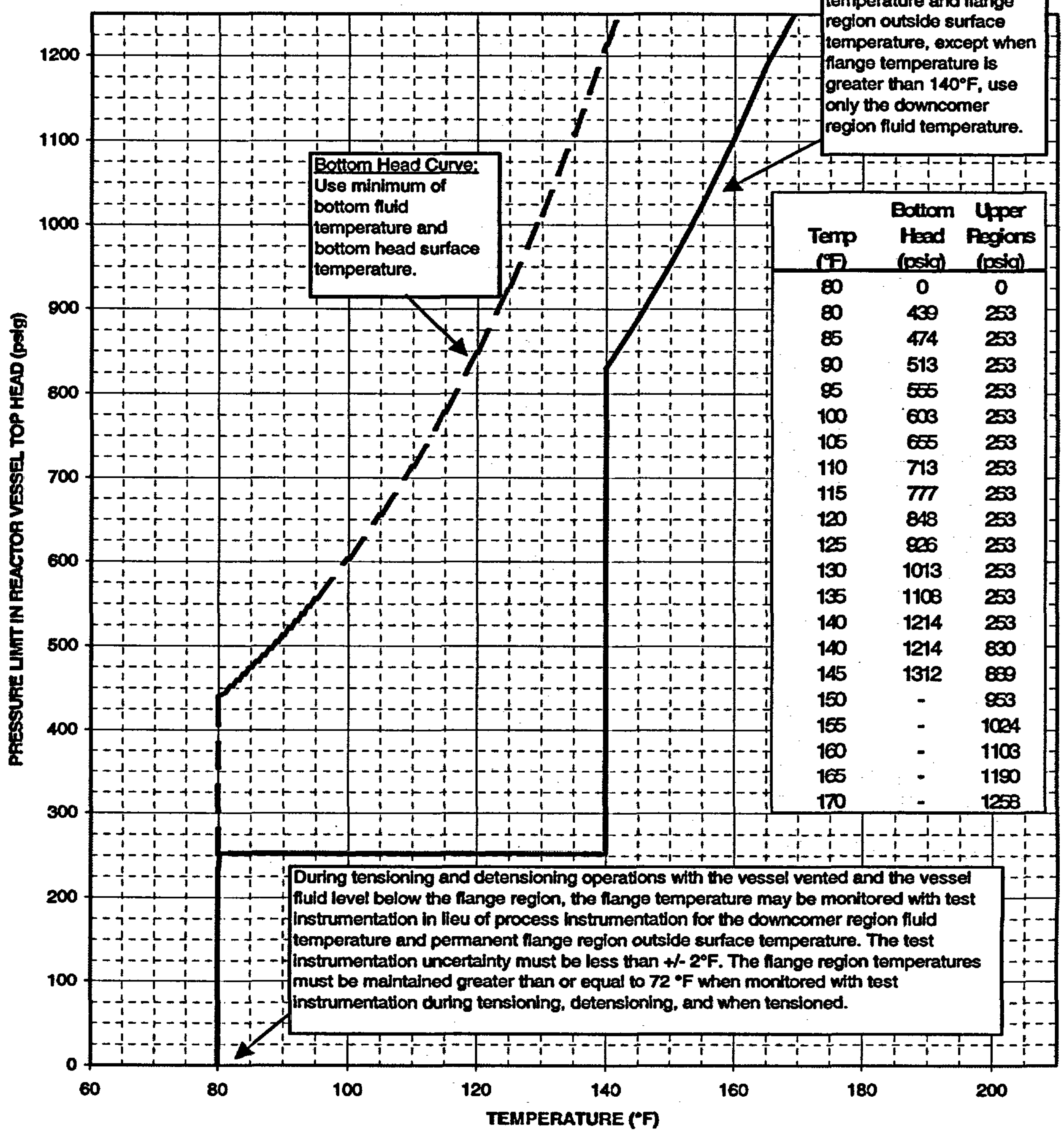


Temperature (°F)	Bottom Head (psig)	Upper Regions (psig)
80	0	0
80	665	253
85	712	253
90	764	253
95	821	253
100	885	253
105	954	253
110	1032	253
110	1032	842
115	1117	885
120	1211	932
125	1316	984
130	-	1042
135	-	1105
140	-	1175
145	-	1253

FIGURE 3.6.2

Reactor Vessel Pressure-Temperature Limitations
Normal Operation, Core Not Critical

100°F/hr Heatup/Cooldown Limit
Valid Through 4.827E8 MWH(t)

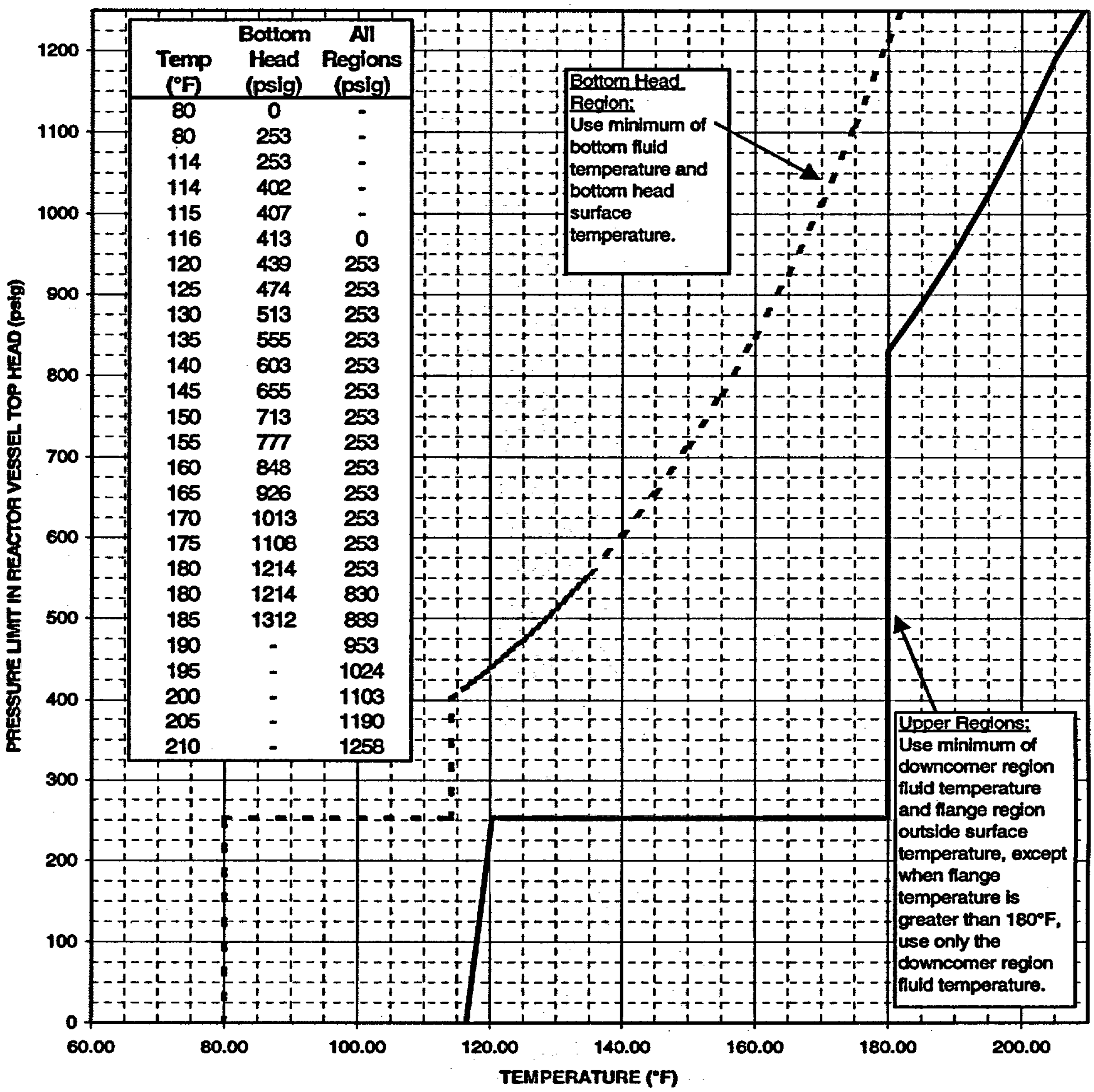


Temp (°F)	Bottom Head (psig)	Upper Regions (psig)
80	0	0
80	439	253
85	474	253
90	513	253
95	555	253
100	603	253
105	655	253
110	713	253
115	777	253
120	848	253
125	926	253
130	1013	253
135	1108	253
140	1214	253
140	1214	830
145	1312	889
150	-	953
155	-	1024
160	-	1103
165	-	1190
170	-	1258

FIGURE 3.6.3

Reactor Vessel Pressure-Temperature Limitations
Normal Operation, Core Critical

100°F/hr Heatup/Cooldown Limit
If Pressure < 253 psig, Water Level must be within
Normal Range for Power Operation
Valid Through 4.827E8 MWH(t)



Temp (°F)	Bottom Head (psig)	All Regions (psig)
80	0	-
80	253	-
114	253	-
114	402	-
115	407	-
116	413	0
120	439	253
125	474	253
130	513	253
135	555	253
140	603	253
145	655	253
150	713	253
155	777	253
160	848	253
165	926	253
170	1013	253
175	1108	253
180	1214	253
180	1214	830
185	1312	889
190	-	953
195	-	1024
200	-	1103
205	-	1190
210	-	1258

BASES:

3.6 and 4.6 REACTOR COOLANT SYSTEM

A. Pressure and Temperature Limitations

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 4.2 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

The Pressure/Temperature (P/T) curves included as Figures 3.6.1, 3.6.2, and 3.6.3 were developed using 10CFR50 Appendix G, 1995 ASME Code, Section XI, Appendix G (including the Summer 1996 Addenda), and ASME Code Case N-640. These three curves provide P/T limit requirements for Pressure Test, Core Not Critical, and Core Critical. The P/T curves are not derived from Design Basis Accident analysis. They are prescribed to avoid encountering pressure, temperature or temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the reactor pressure boundary, a condition that is unanalyzed.

During heating events, the thermal gradients in the reactor vessel wall produce thermal stresses that vary from compressive at the inner wall to tensile at the outer wall. During cooling events the thermal stresses vary from tensile at the inner wall to compressive at the outer wall. The thermally induced tensile stresses are additive to the pressure induced tensile stresses. In the flange region, bolt preload has a significant affect on stress in the flange and adjacent plates. Therefore heating/cooling events and bolt preload are used in the determination of the pressure-temperature limitations for the vessel.

The guidance of Branch Technical Position - MTEB 5-2, material drop weight, and Charpy impact test results were used to determine a reference nil-ductility temperature (RT_{NDT}) for all pressure boundary components. For the plates and welds adjacent to the core, fast neutron (E > 1 Mev) irradiation will cause an increase in the RT_{NDT}. For these plates and welds an adjusted RT_{NDT} (ART_{NDT}) of 89°F and 73°F (¼ and ¾ thickness locations) was conservatively used in development of these curves for core region components. Based upon plate and weld chemistry, initial RT_{NDT} values, predicted peak fast neutron fluence (3.18×10^{17} n/cm^2 at the reactor vessel inside surface) for a gross power generation of 4.827×10^8 MWH(t), these core region ART_{NDT} values conservatively bound the guidance of Regulatory Guide 1.99, Revision 2.

There were five regions of the reactor pressure vessel (RPV) that were evaluated in the development of the P/T Limit curves: (1) the reactor vessel beltline region, (2) the bottom head region, (3) the feedwater nozzle, (4) the recirculation inlet nozzle, and (5) the upper vessel flange region. These regions will bound all other regions in the vessel with respect to considerations for brittle fracture.

Two lines are shown on each P/T limit figure. The dashed line is the Bottom Head Curve. This is applicable to the bottom head area only and includes the bottom head knuckle plates and dollar plates. Based on bottom head fluid temperature and bottom head surface temperature, the reactor pressure shall be maintained below the dashed line at all times.

BASES: 3.6 and 4.6 (Cont'd)

C. Coolant Leakage

The 5 gpm limit for unidentified leaks was established assuming such leakage was coming from the reactor coolant system. Tests have been conducted which demonstrate that a relationship exists between the size of a crack and the probability that the crack will propagate. These tests suggest that for leakage somewhat greater than the limit specified for unidentified leakage, the probability is small that imperfections or cracks associated with such leakage would grow rapidly. Leakage less than the limit specified can be detected within a few hours utilizing the available leakage detection systems. If the limit is exceeded and the origin cannot be determined in a reasonably short time the plant should be shutdown to allow further investigation and corrective action.

The 2 gpm increase limit in any 24 hour period for unidentified leaks was established as an additional requirement to the 5 gpm limit by Generic Letter 88-01, "NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping."

The removal capacity from the drywell floor drain sump and the equipment drain sump is 50 gpm each. Removal of 50 gpm from either of these sumps can be accomplished with considerable margin.

D. Safety and Relief Valves

Safety analyses have shown that only three of the four relief valves are required to ensure compliance with the MCPR safety limit for the analyzed transients. The setpoint tolerance value for as-left or refurbished valves is specified in Section III of the ASME Boiler and Pressure Vessel Code as ±1% of set pressure. However, the code allows a larger tolerance value for the as-found condition if the supporting design analyses demonstrate that the applicable acceptance criteria are met. For the purposes of this limiting condition, a relief valve that is unable to actuate within tolerance of its set pressure is considered to be as inoperable as a mechanically malfunctioning valve. Safety analysis has been performed which shows that with all safety and safety relief valves within ±3% of the specified set pressures in Table 2.2.1 and with one inoperable safety relief valve, the reactor coolant pressure safety limit of 1375 psig and the MCPR safety limit are not exceeded during the limiting overpressure transient.

3.11 LIMITING CONDITIONS FOR OPERATION

3.11 REACTOR FUEL ASSEMBLIES

Applicability:

The Limiting Conditions for Operation associated with the fuel rods apply to these parameters which monitor the fuel rod operating conditions.

Objective:

The Objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.

Specifications:

A. Average Planar Linear Heat Generation Rate (APLHGR)

During operation at $\geq$23% Rated Thermal Power, the APLHGR for each type of fuel as a function of average planar exposure, power, and flow shall not exceed the limiting values provided in the Core Operating Limits Report. For single recirculation loop operation, the limiting values shall be the values provided in the Core Operating Limits Report listed under the heading "Single Loop Operation." If at any time during operation at $\geq$23% Rated Thermal Power it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, APLHGR(s) shall be returned to within prescribed limits within two (2) hours; otherwise, the reactor shall be brought to <23% Rated Thermal Power within 4 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

4.11 SURVEILLANCE REQUIREMENTS

4.11 REACTOR FUEL ASSEMBLIES

Applicability:

The Surveillance Requirements apply to the parameters which monitor the fuel rod operating conditions.

Objective:

The Objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.

Specifications:

A. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure, power, and flow shall be determined once within 12 hours after $\geq$23% Rated Thermal Power and daily during operation at $\geq$23% Rated Thermal Power thereafter.

3.11 LIMITING CONDITIONS FOR OPERATION

B. Linear Heat Generation Rate (LHGR)

During operation at ≥23% Rated Thermal Power, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the maximum allowable LHGR provided in the Core Operating Limits Report.

If at any time during operation at ≥23% Rated Thermal Power it is determined by normal surveillance that the limiting value for LHGR is being exceeded, LHGR(s) shall be returned to within the prescribed limits within two (2) hours; otherwise, the reactor shall be brought to <23% Rated Thermal Power within 4 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

4.11 SURVEILLANCE REQUIREMENTS

B. Linear Heat Generation Rate (LHGR)

The LHGR as a function of core height shall be checked once within 12 hours after ≥23% Rated Thermal Power and daily during operation at ≥23% Rated Thermal Power thereafter.

3.11 LIMITING CONDITIONS FOR OPERATION

C. Minimum Critical Power Ratio (MCPR)

1. During operation at ≥23% Rated Thermal Power the MCPR operating value shall be equal to or greater than the MCPR limits provided in the Core Operating Limits Report. For single recirculation loop operation, the MCPR Limits at rated flow are also provided in the Core Operating Limits Report. If at any time during operation at ≥23% Rated Thermal Power it is determined by normal surveillance that the limiting value for MCPR is being exceeded, MCPR(s) shall be returned to within the prescribed limits within two (2) hours; otherwise, the reactor power shall be brought to <23% Rated Thermal Power within 4 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

4.11 SURVEILLANCE REQUIREMENTS

C. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined once within 12 hours after ≥23% Rated Thermal Power and daily during operation at ≥23% Rated Thermal Power thereafter.

BASES:

4.11 FUEL RODS

A. The APLHGR, LHGR and MCPR shall be checked daily when operating at ≥23% Rated Thermal Power to determine if fuel burnup, or control rod movement has caused changes in power distribution. Since changes due to burnup are slow, and only a few control rods are removed daily, a daily check of power distribution is adequate. For a limiting value to occur below 23% of rated thermal power, an unreasonably large peaking factor would be required, which is not the case for operating control rod sequences. The 12 hour allowance after thermal power ≥23% Rated Thermal Power is achieved is acceptable given the large inherent margin to operating limits at low power levels.

B. At certain times during plant startups and power changes the plant technical staff may determine that surveillance of APLHGR, LHGR and/or MCPR is necessary more frequently than daily. Because the necessity for such an augmented surveillance program is a function of a number of interrelated parameters, a reasonable program can only be determined on a case-by-case basis by the plant technical staff. The check of APLHGR, LHGR and MCPR will normally be done using the plant process computer. In the event that the computer is unavailable, the check will consist of either a manual calculation or a comparison of existing core conditions to those existing at the time of a previous check to determine if a significant change has occurred.

If a reactor power distribution limit is exceeded, an assumption regarding an initial condition of the DBA analysis, transient analyses, or the fuel design analysis may not be met. Therefore, prompt action should be taken to restore the APLHGR, LHGR or MCPR to within the required limits such that the plant operates within analyzed conditions and within design limits of the fuel rods. The 2 hour completion time is sufficient to restore the APLHGR, LHGR, or MCPR to within its limits and is acceptable based on the low probability of a transient or DBA occurring simultaneously with the APLHGR, LHGR, or MCPR out of specification.

C. Minimum Critical Power Ratio (MCPR) - Surveillance Requirement

At core thermal power levels less than or equal to 23%, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. During initial start-up testing of the plant, a MCPR evaluation will be made at 23% thermal power level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR above 23% rated thermal power is sufficient since power distribution shifts are very slow during normal operation.