GE Nuclear Energy

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LICENSING TOPICAL REPORT

CONSTANT PRESSURE POWER UPRATE

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INFORMATION NOTICE

This is a non-proprietary version of the document NEDC-33004P, which has the proprietary information removed. Portions of the document that have been removed are indicated by an open and closed bracket as shown here [ ].

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

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EXECUTIVE SUMMARY

GE has previously developed and implemented Extended Power Uprate. Based on the Extended Power Uprate experience, GE has developed an approach to uprate reactor power that maintains the current plant reactor dome pressure. By performing the power uprate with no pressure increase, the effect on the plant safety analyses and system performance is reduced, thus allowing for a more streamlined process.

This report provides a systematic disposition of the engineering assessments required to support a Constant Pressure Power Uprate. These dispositions include generic assessments that are based on both analysis and experience with Extended Power Uprate projects previously provided through specific plant submittals.

To further ease future NRC reviews, a prescribed approach to be used for each plant specific power uprate submittal is also provided. Future plant specific submittals of Constant Pressure Power Uprate will include a plant specific document based on the approach prescribed herein consistent with the dispositions documented in this report.

REVISIONS

NEDC-33004P, Revision 0 was submitted for NRC review on March 20, 2001. Feedback received, following an initial NRC staff review, has been factored into this revision of NEDC-33004P, Revision 1. The key changes in Revision 1 are reduction and reclassification of the disposition categories and revised plant specific submittal requirements. NEDC-33004P, Revision 1 replaces NEDC-33004P, Revision 0 in its entirety and should be the sole basis for NRC review and approval.

NEDC-33004P, Revision 1, including Errata & Addenda 1, was submitted for NRC review on December 21, 2001. The revision included modifications in response to NRC RAIs and clarifications as noted with the December 21, 2001 transmittal letter.

Revision 3 is based on the above December 21, 2001 version. The key changes included in Revision 3 are: (1) modifications of Section 1. to clarify the understanding with the NRC regarding licensing changes concurrent with a CLTR based power uprate, and (2) the re-marking of the proprietary content consistent with the modified affidavit. The changes contained in Errata & Addenda 2, as incorporated into Revision 2, have been removed from Revision 3.

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ACRONYMS

Acronym	Definition
AC	Alternating Current
ADS	Automatic Depressurization System
ALARA	As Low As Reasonably Achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
AOP	Abnormal Operating Procedure
APLHGR	Average Planner Linear Heat Generation Rate
APRM	Average Power Range Monitor
ART	Adjusted Reference Temperature
ARTS	Average Power Range Monitor, Rod Block Monitor, Technical Specifications Improvement Program
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
BHP	Brake Horse Power
BOP	Balance of Plant
BWR	Boiling Water Reactor
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CLTP	Current Licensed Thermal Power
COLR	Core Operating Limits Report
CPPU	Constant Pressure Power Uprate
CRD	Control Rod Drive
CRDA	Control Rod Drop Accident
CS	Core Spray
CSC	Containment Spray Cooling
CSS	Core Support Structure
DBA	Design Basis Accident
DC	Direct Current

Acronym	Definition
E1A	Stability Enhanced Option 1A
ECCS	Emergency Core Cooling System
ELTR 1	NEDC-32424P-A (Reference 1)
ELTR 2	NEDC-32523P-A (Reference 2)
EMA	Equivalent Margin Analysis
EOC	End of Cycle
EOP	Emergency Operating Procedures
EPU	Extended Power Uprate
EQ	Environmental Qualification
FAC	Flow-Accelerated Corrosion
FCV	Flow Control Valve
FHA	Fuel Handing Accident
FIV	Flow-Induced Vibration
FPCC	Fuel Pool Cooling and Cleanup
FW	Feedwater
GNF	Global Nuclear Fuel
HELB	High Energy Line Break
HEPA	High Efficiency Particulate Air
HPCI	High Pressure Coolant Injection
HPCS	High Pressure Core Spray
HPSP	High Pressure Setpoint
HPT	High Pressure Turbine
HVAC	Heating, Ventilation and Air Conditioning
IASCC	Irradiation Assisted Stress Corrosion Cracking
IC	Isolation Condenser
IEEE	Institute of Electrical and Electronic Engineers
ILBA	Instrument Line Break Accident
IORV	Inadvertent Opening of Relief Valve
IPE	Individual Plant Evaluation
IRM	Intermediate Range Monitor
ITS	Improved Technical Specification

Acronym	**Definition**
LERF	Large Early Release Fraction
LHGR	Linear Heat Generation Rate
LOCA	Loss-of-Coolant Accident
LOOP	Loss of Offsite Power
LPCI	Low Pressure Coolant Injection
LPCS	Low Pressure Coolant Spray
LPRM	Local Power Range Monitors
LPSP	Low Power Setpoint
LTR	Licensing Topical Report
MAPLHGR	Maximum Average Planer Linear Heat Generation Rate
MCPR	Minimum Critical Power Ratio
MELB	Moderate Energy Line Break
MELLLA	Maximum Extended Load Line Limit Analysis
MEOD	Maximum Extended Operating Domain
M-G	Motor-Generator
MSIV	Main Steam Isolation Valve
MSIVC	Main Steam Isolation Valve Closure
MSIVF	Main Steam Isolation Valve Closure with Scram on High Neutron Flux
MSL	Main Steam Line
MSLBA	Main Steam Line Break Accident
NEMA	National Electric Manufactures' Association
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OLTP	Original Licensed Thermal Power
OPRM	Oscillation Power Range Monitor
P/T	Pressure-Temperature
PCS	Pressure Control System
PCT	Peak Cladding Temperature
PEA	Primary Element Accuracy
PMA	Process Monitoring Accuracy

Acronym	Definition
PRA	Probability Risk Assessment
PRFO	Pressure Regulator Failure-Open
PUSAR	Power Uprate Safety Analysis Report
RBM	Rod Block Monitor
RCIC	Reactor Core Isolation Cooling
RCIS	Rod Control and Information System
RCPB	Reactor Coolant Pressure Boundary
RHR	Residual Heat Removal
RIPDs	Reactor Internal Pressure Differences
RPC	Rod Pattern Controller
RPS	Reactor Protection System
RPT	Recirculation Pump Trip
RPV	Reactor Pressure Vessel
RRC	Reactor Recirculation Coolant
RTP	Rated Thermal Power
RWCU	Reactor Water Cleanup
RWE	Rod Withdrawal Error
RWL	Rod Withdrawal Limiter
RWM	Rod Worth Minimizer
SAFER/GESTR-LOCA	A computer program, a vessel blowdown model for analysis of system response to loss-of-coolant accident
SAR	Safety Analysis Report
SC	Steam Condensing
SER	Safety Evaluation Report
SBO	Station Blackout
SDC	Shutdown Cooling
SGTS	Standby Gas Treatment System
SJAE	Steam Jet Air Ejector
SLC	Standby Liquid Control
SLCS	Standby Liquid Control System
SLMCPR	Safety Limit Minimum Critical Power Ratio
SLO	Single-Loop Operation

Acronym	**Definition**
SPC	Suppression Pool Cooling
SPU	Stretch Power Uprate
SRLR	Supplemental Reload Licensing Report
SRM	Source Range Monitor
SRP	Standard Review Plan
SRV	Safety Relief Valve
TCV	Turbine Control Valve
T-G	Turbine Generator
TIP	Traversing Incore Probe
UFSAR	Updated Final Safety Analysis Report
UHS	Ultimate Heat Sink
USE	Upper Shelf Energy

1.0 INTRODUCTION

Previously, General Electric (GE) submitted a set of generic guidelines to be met and a general approach to be followed for plants that extended reactor thermal power up to 120% of their original licensed thermal power. These guidelines and subsequent evaluations were based on the assumption that the maximum operating reactor pressure also would be increased. These guidelines and evaluations, together with associated Nuclear Regulatory Commission (NRC) position and Safety Evaluation Reports, are provided in References 1 and 2 (ELTR 1/2) and have been applied to all extended power uprate submittals since their NRC approval.

Subsequent to the submittal of these licensing topical reports for approval, GE has developed a different approach to uprating reactor power. This approach maintains the current plant maximum operating reactor pressure. The power uprate with no pressure increase has been utilized at several plants and will be pursued for most of the future power uprate applications. GE's current experience base with power uprate is provided in Table 1-1. By performing the power uprate with no pressure increase, there is a substantially smaller effect on the plant safety analysis and system performance. This constraint allows a more streamlined approach to power uprate analyses and evaluations.

The purpose of this Licensing Topical Report (LTR) is to document the approach to be followed and provide the basis for future Constant Pressure Power Uprate (CPPU) applications. The overall approach has been streamlined consistent with the constant pressure assumption. In addition, experience with previous power uprate applications, new generic evaluations, and the standard reload analysis process, have been factored into the overall approach to simplify the required plant specific documentation while maintaining a rigorous and systematic licensing and safety evaluation. Further, the focus of the evaluation has been placed on the safety evaluations required for power uprate to allow for a more comprehensive and streamlined review process.

For this report, it is assumed that the only change to the plant licensing and design basis is an increase of up to 20% in the plant 100% Original Licensed Thermal Power (OLTP). The CPPU approach generically dispositions, defers to the standard reload or fuel introduction process, simplifies, or limits some of the safety analyses and system performance evaluations used to support operation at the higher power level. GE has been informed by the NRC and agrees that Licensees proposing to reference this topical report as a basis for a power uprate license amendment request, and proposing to obtain a license amendment to incorporate one or more of the plant changes listed below must first request and obtain a license for the associated change prior to the start of the staff review of the power uprate request that references this topical report. The one exception is with regards to a source term methodology change. GE has been informed by the NRC that a Licensee may submit and the staff will review a source term methodology change, in lieu of the analysis in Section 9.2 of this report, concurrent with the power uprate request, if the source term submittal supports operation at the uprated power level.

- No change in the current maximum normal operating reactor dome pressure,

[

]

The CPPU analyses and evaluations provided in the plant specific submittal will be performed consistent with the intended licensing basis of the plant as it will operate after implementation of the power uprate, including all previously submitted and approved license amendment requests. The CPPU operating map is an extension of the current ARTS/MELLLA or MEOD operating map. Therefore, this report is applicable only to plants that are licensed, to operate with the ARTS/MELLLA or MEOD operational margin improvement option. A typical power/flow map showing the CPPU change in applicable operating conditions is shown on Figure 1-1.

Changes to the plant licensing and design basis necessary to support the licensing of the power uprate will be reported and justified in a plant specific power uprate submittal. The plant specific submittal will include changes to the analysis basis methodology identified in References 1 and 2, unless this methodology is revised by this report. Applicable new methods that are approved by the NRC independent of this LTR may be used after they are approved by the staff. GE has been informed by the NRC and agrees that any new methods that a licensee wishes to have reviewed or implement concurrent with the CPPU approach may cause the NRC staff, at their sole discretion, to determine that the generic disposition of any analysis or evaluation in this LTR is no longer valid, or that the scope of the submitted plant specific evaluations is inadequate, and may require the submittal of substantial additional supporting analyses and evaluations during the review of that application, which may substantially extend the review scope and schedule.

Because of the reduced effect of a CPPU on many safety evaluations, a number of generic evaluations are provided to support the plant specific submittals. In addition, some generic assessments from References 1 and 2 can be utilized because they bound the effect of the CPPU approach. This report provides the results of these evaluations, assessments, and dispositions for NRC approval, thus simplifying the plant specific NRC review required for each new CPPU submittal.

To further simplify future NRC reviews of plant specific CPPU submittals, the format of the Power Uprate Safety Analysis Report (PUSAR) to be used for each plant specific CPPU submittal will be based on the format of this report. The PUSAR is based on the above assumptions and includes consideration of the evaluations, assessments, and dispositions provided in this report. Any deviations from the bases and evaluations provided in this report will be included and justified in the plant specific submittal and will be summarized in Section 1 of the plant specific submittal. The level of information to be provided for each plant specific submittal and the format for providing that information will be consistent with past extended power uprate submittals. For those analyses and evaluations that are generically dispositioned in this report, the plant specific PUSAR is only required to provide the basis for the generic dispositions and confirm the applicability of these generic dispositions for the specific plant application. However, GE has been informed by the NRC and agrees that if any plant seeks a concurrent review or implementation of a power uprate and any of the excluded plant changes listed above, the NRC staff may, at their sole discretion, determine that the generic disposition of any analysis or evaluation in this LTR is no longer valid, and may require the submittal of substantial additional supporting analyses and evaluations during the review of that application.

The sections in this topical report that are related to reactor systems and fuel performance are not applicable to, and cannot be referenced by, any plant that (1) is not operating with GE fuel up through GE14, or (2) does not intend to use approved GE analytical methods to perform the reload analyses-of-record supporting plant operation at the uprated power level.

In this LTR, the acronym for an assessment or equipment name is typically provided with the first use of the name (a table of acronyms is provided).

1.1 REPORT APPROACH

The report sections correspond to those previously used on plant specific, extended power uprate submittals. Each of the evaluations included in those submittals have been reviewed and assigned one of the two disposition categories:

- Generic assessment
- Plant Specific evaluation

Each top level section of this report begins with a summary disposition table for all of the principal evaluations included in the section. A principal evaluation is a thermal-hydraulic, nuclear, mechanical (e.g., vessel integrity), or system design (e.g., ECCS) analysis or evaluation that is potentially limiting with respect to safety considerations relative to power uprate. Each principal evaluation is included in a separate subsection, which includes a table with the following information:

- Evaluation topic
- Primary effect of CPPU on topic
- Disposition category for the assessment

The justification of the categorization is included after the table. This justification includes current experience with extended power uprate and the basis for the disposition, as applicable.

The technical dispositions are contained in Sections 2 through 10. General information has also been provided in Section 11 to support utility licensing documentation required for the plant specific CPPU submittal. This general information provides a template to the utility for development of the environmental report, plant technical specification changes, and significant hazards assessment. This information is provided for use by the utility, and NRC review is only requested for the level of detail presented. The utility may elect to reference some or all of the information given in Section 11 in the documentation supporting the plant specific licensing CPPU submittal.

The term "Constant Pressure Power Uprate" refers, in this report, to the general approach for power uprate outlined above, including all disposition categories and the exclusions identified in Section 1.0.

1.1.1 Generic Assessments

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 upon either a demonstration that previous pressure increase power uprate assessments provided in Reference 1 or 2 are bounding or upon specific generic studies provided for the CPPU. For these bounding analyses, the current CPPU experience is provided along with the basis and results of the assessment. If the generic assessment is fuel design dependent, this assessment is applicable only to GE/GNF fuel designs up through GE 14, analyzed with GE methodology. The effect of CPPU on future GE/GNF fuel designs is addressed during the assessment of the new fuel design consistent with the requirements of Reference 3.

For those CPPU assessments having a negligible effect, the current CPPU experience plus a phenomenological discussion of the basis for the assessment is provided. Reference 1 or 2 is referenced if the information in these reports supports the conclusion of negligible effect. Any plant system design that falls outside of the current experience base for a generic analysis will be addressed in the plant specific submittal.

Some of the safety evaluations affected by CPPU are fuel operating cycle (reload) dependent. Reload dependent evaluations require that the reload fuel design, core loading pattern, and operational plan be established so that analyses can be performed to establish core operating limits. The reload analysis demonstrates that the core design for CPPU meets the applicable NRC evaluation criteria and limits documented in Reference 3. [

] Therefore, the reload fuel design and core loading pattern dependent plant evaluations for CPPU operation will be performed with the reload analysis as part of the standard reload licensing process. No plant can implement a power uprate unless the appropriate reload core analysis is performed and all criteria and limits documented in Reference 3 are satisfied. Otherwise, the plant would be in an unanalyzed condition. Based on current requirements, the reload analysis results are documented in the Supplemental Reload Licensing Report (SRLR), and the applicable core operating limits are documented in the plant specific Core Operating Limits Report (COLR).

Generic dispositions for reload analysis assessments are described in the appropriate sections of this report. For these assessments, a phenomenological discussion of the effect of CPPU on the expected analysis results is provided along with the relative experience base and reference to supporting information provided by either Reference 1 or 2.

The applicability of the generic assessments for a specific plant application will be evaluated. The plant specific submittal will either document the successful confirmation of the generic assessment or provide a plant specific evaluation, consistent with Section 1.1.2, if the applicability assessment is unsuccessful. However, GE has been informed by the NRC and agrees that if any plant seeks a concurrent review or implementation of a power uprate and any of the excluded plant changes described in Section 1.0, the NRC staff may, at their sole discretion, determine that the generic disposition of any analysis or evaluation in this LTR is no longer valid, and may require the submittal of substantial additional supporting analyses and evaluations during the review of that application.

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, 2 or 3, these documents are referenced rather than the original report.

The plant specific evaluations will be reported in the plant specific submittal consistent with the level of detail of previous extended power uprate submittals or as indicated in this report.

However, GE has been informed by the NRC and agrees that if any plant seeks a concurrent review or implementation of a power uprate and any of the excluded plant changes listed above, the NRC staff may, at their sole discretion, determine that the generic disposition of any analysis or evaluation in this LTR is no longer valid, or that the scope of the submitted plant specific evaluations is inadequate, and may require the submittal of substantial additional supporting analyses and evaluations during the review of that application.

1.2 EFFECT OF CPPU

1.2.1 Operating Domain

The upper bound of the operating domain is defined by the current MELLLA/MEOD upper boundary. The MELLLA/MEOD upper boundary remains unchanged with CPPU in terms of absolute power and core flow, and is extended up to the new 100% core power value. [

] The effect of CPPU on the other power flow map boundaries is provided in Table 1-2. No other changes in the plant operational flexibility options that affect the operating domain are assumed, as noted in Section 1.0.

1.2.2 Nuclear and Thermal-Hydraulic Evaluations

The change in the power level will affect the plant steady-state heat balance. The typical effect of a 20% increase in reactor power on plant operating parameters is shown in Table 1-3. This table shows the average change and range of heat balance parameter values for representative BWRs over the range of plant sizes and product lines. These results show that the effect of a 20% increase in power with no reactor pressure increase across the BWR fleet is fairly uniform. The plant specific submittal will include a summary of steady state parameters based on the plant specific CPPU heat balance.

Experience has demonstrated that CPPU may have an effect on thermal-hydraulic safety analyses. [

] Several of the other thermal-hydraulic safety analyses can be performed on a generic basis, and the results are documented in this report. The remaining thermal-hydraulic safety analyses require plant specific evaluations. The plant specific evaluation or applicability confirmation will be provided in the plant specific submittal, as applicable.

The nuclear evaluation requirements and criteria for the limits are not changed as a result of CPPU. The shutdown margin and hot excess reactivity requirements identified in Reference 3 remain applicable. CPPU increases the average power density proportional to the power increase and has some effects on the core operating and design flexibility, reactivity characteristics and energy requirements. No changes in the fuel mechanical designs or fuel design limits are required to implement CPPU. The additional energy requirements for power uprate are met by an increase in bundle enrichment, an increase in reload batch size, and/or changes in fuel loading pattern to maintain the desired plant operating cycle length. The power distribution in the core is established to achieve increased core power while satisfying the core operating limits. [

]

1.2.3 Mechanical Evaluations

The primary effects that require evaluation for mechanical components are an increase in fluence, reactor internal pressure differences (RIPDs), flow and temperature. Certain evaluations for the mechanical components are performed on a generic basis. However, there are some significant plant specific evaluations that are required. Increased fluence results in increased embrittlement of the reactor pressure vessel (RPV) requiring a plant specific evaluation. An increase in feedwater (FW) flow and temperature will result in an increase of stress and fatigue of the FW nozzle also [
] For reactor internals, it is expected that the existing/original design basis loads bounds the CPPU loads; [
] For example, an increase in RIPDs results in increased stress and fatigue of RPV internals, including the shroud attachment to the RPV. Increased flow rates of the main steam and FW result in increased vibration of piping; a vibration test program is recommended for these piping components. Flow-induced vibration of the RPV internals will be evaluated [ ] The increase in flow and temperature of the FW and main steam line (MSL) piping will require [
]

1.2.4 System Evaluations

Experience has demonstrated that the effect of CPPU on Nuclear Steam Supply System (NSSS) and Balance Of Plant (BOP) systems is system dependent. Overall, many NSSS and BOP systems are not significantly challenged by CPPU. Where appropriate, a generic disposition is provided for systems that are not significantly affected by CPPU.

For the remainder of the NSSS and BOP safety systems, there is typically sufficient capability that no system modifications are required. This capability is demonstrated by system-specific evaluations. If modifications are required to meet safety requirements, this will be noted in the plant specific submittals.

For BOP power generation systems required for normal operation, modifications (e.g., new turbine rotating elements and condensate or feedwater pump modifications) are typically required to accommodate the increased steam and feedwater flow. These modifications typically affect non-safety related power generating and supporting systems.

Limited Technical Specification setpoint changes are required as a result of CPPU. Typically, setpoint changes are limited to the Neutron Monitoring System, main steamline high flow, and turbine first-stage pressure.

Table 1-1 GE Power Uprate Experience

Plant	Stretch/Extended Power Uprate	Uprate Power (~ % OLTP)	Reactor Dome Pressure Increased
Duane Arnold	SPU	105	Yes
Cofrentes	SPU	105	Yes
Hatch - 1, 2	SPU	105	Yes
Susquehanna - 1, 2	SPU	105	Yes
WNP-2	SPU	105	Yes
Limerick - 1, 2	SPU	105	Yes
Peach Bottom - 2, 3	SPU	105	Yes
Fermi 2	SPU	105	Yes
FitzPatrick	SPU	105	Yes
Brunswick - 1, 2	SPU	105	Yes
NMP-2	SPU	105	Yes
Browns Ferry - 2, 3	SPU	105	Yes
River Bend	SPU	105	Yes
KKM	EPU	114	Yes
KKL	EPU	117	Yes
Laguna Verde - 1, 2	SPU	105	No
LaSalle - 1, 2	SPU	105	No
Perry	SPU	105	No
Hatch - 1, 2	EPU	113	No
Monticello	EPU	106	No
Cofrentes *	EPU	110	No
Duane Arnold *	EPU	120	No
Dresden - 2, 3 *	EPU	117	No
Quad Cities - 1, 2 *	EPU	117	No
Clinton *	EPU	120	No
Brunswick - 1, 2 *	EPU	120	No
Browns Ferry 2, 3 *	EPU	120	No

*In progress.

Table 1-2 Effect of 20% Power Uprate on Power Flow Map Boundaries

[

]

Table 1-3 Change in Plant Operating Parameters for a 20% Increase in Core Thermal Power

[

]

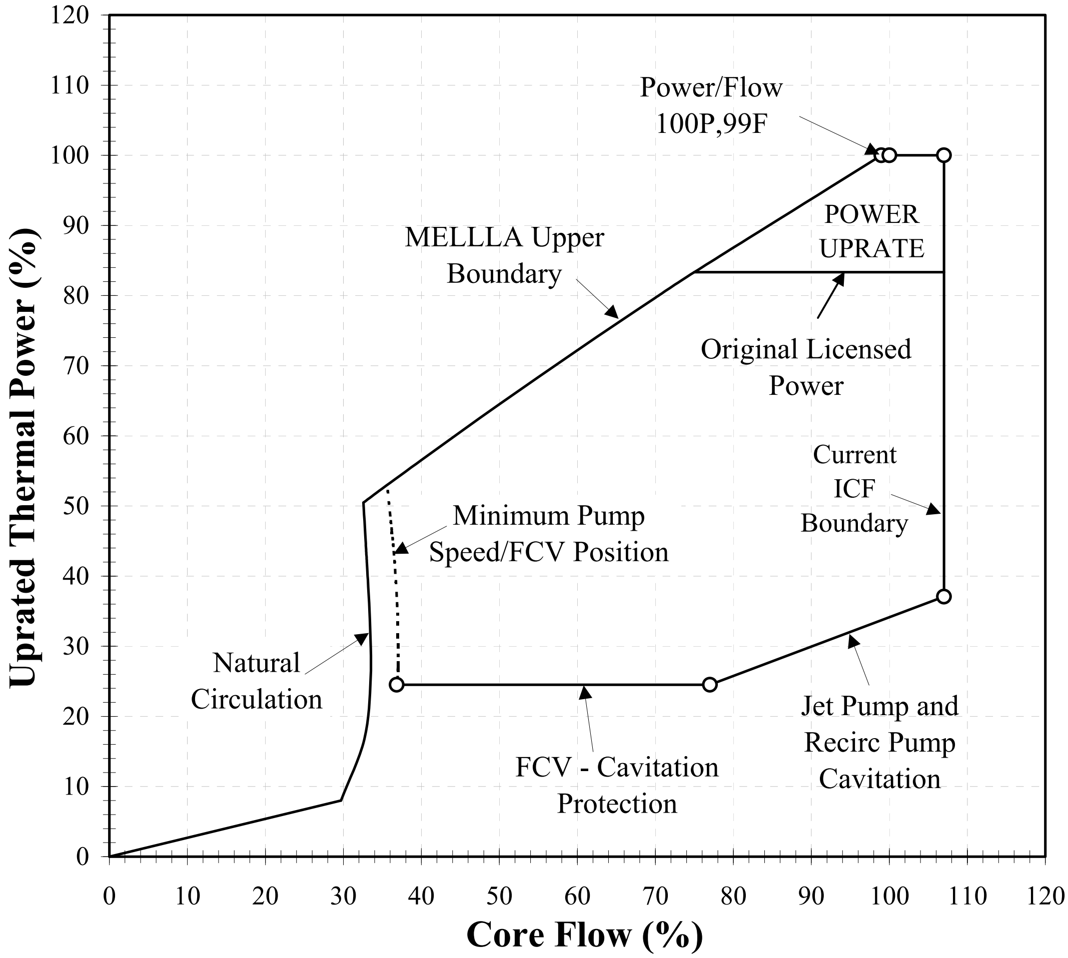

Figure 1-1. Typical CPPU-Based Power Uprate Power/Flow Map

2.0 REACTOR CORE AND FUEL PERFORMANCE

This section addresses the evaluations in Regulatory Guide 1.70, Chapter 4, that are documented in the current plant extended power uprate submittals. The major evaluations and summary disposition of these evaluations are as follows:

Section	Title	Generic	Plant Specific
2.1	Fuel Design and Operation	[	
2.2	Thermal Limit Assessment		
2.3	Reactivity Characteristics		
2.4	Stability		
2.5	Reactivity Control		]

The detailed assessment dispositions as outlined in Section 1.1 are provided in the applicable sections. The plant specific evaluations will be reported in the plant specific submittal consistent with the format and level of detail of previous extended power uprate submittals or as indicated below. The applicability of the generic assessments for a specific plant application will be evaluated. The plant specific submittal will either document the successful confirmation of the generic assessment or provide a plant specific evaluation if the applicability assessment is unsuccessful.

2.1 FUEL DESIGN AND OPERATION

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

Topic	CPPU Effect	Disposition
Fuel product line design	None	[
Core design	Increased average power density	
Fuel thermal margin monitoring threshold	Increased average power density	]

CPPU Effect: [ ] Power uprate requires an increase in the energy loaded into the core each cycle.

CPPU Basis: CPPU increases the average power density proportional to the power increase and has some effects on operating flexibility, reactivity characteristics and energy requirements. The maximum allowable peak bundle power is not increased by power uprate. The additional energy requirements for power uprate 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 MCPR, LHGR, and MAPLHGR in any individual fuel bundle to be within its allowable value as defined in the COLR.

[

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

]

The 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. [

]

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

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

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 is addressed below. The topics considered in this section are:

Topic	CPPU Effect	Disposition
2.2.1 Safety Limit MCPR	Flatter radial power distribution	[
2.2.2 MCPR Operating Limit	Little effect	
2.2.3 MAPLHGR Limit	Little or no effect	
2.2.3 Maximum LHGR Limit	No effect	]

2.2.1 Safety Limit MCPR

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

CPPU Basis: [ ] This effect is not changed by following the constant pressure approach for the uprate. The SLMCPR analysis reflects the actual plant core loading pattern and is performed for each plant reload core (see Reference 3). [ ]

2.2.2 MCPR Operating Limit

CPPU Effect: CPPU operating conditions [ ]

CPPU Basis: The MCPR Operating Limit is calculated by adding the change in MCPR due to the limiting Anticipated Operational Occurrence (AOO) event to the SLMCPR and is determined on a cycle specific basis. Power uprate does not change the method used to determine this limit. The effect of power uprate on AOO events is addressed in Section 9.1. [ ] This remains valid for uprates performed at constant dome operating pressure. [ ]

2.2.3 MAPLHGR and Maximum LHGR Operating Limits

CPPU Effect: CPPU operating conditions do not usually affect the MAPLHGR or LHGR operating limits.

CPPU Basis: The MAPLHGR and LHGR limits ensure that the plant does not exceed regulatory limits established in 10CFR50.46 or by the fuel design limits. The MAPLHGR Operating Limit is determined by analyzing the limiting loss-of-coolant accident (LOCA) for the plant. As discussed in Section 4.3, [ ] The maximum LHGR limit is determined by the fuel rod thermal-mechanical design and is not affected by CPPU. [ ] For older core monitoring systems, the LHGR limits are combined with the MAPLHGR limits. For newer core monitoring systems, the LHGR and MAPLHGR are monitored directly for compliance with the fuel thermal-mechanical operating limits and LOCA limits, respectively.

2.3 REACTIVITY CHARACTERISTICS

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

Topic	CPPU Effect	Disposition
Hot excess reactivity	May lower excess reactivity	[
Shutdown margin	May reduced shutdown margin	]

CPPU Effect: The higher core energy requirements of power uprate may reduce the hot excess reactivity and reduce operating shutdown margins.

CPPU Basis: The general effect of power uprate on core reactivity is described in Section 5.7.1 of Reference 1, and is also applicable for power uprate with no pressure increase. Based on experience with many previous plant specific power uprate submittals, the required hot excess reactivity and shutdown margin can be achieved for power uprates through appropriate fuel and core design. [ ]plant shutdown and reactivity margins must meet NRC approved limits established in Reference 3 on a cycle specific basis and are evaluated for each plant reload core[
]

2.4 STABILITY

Section 3.2 of Reference 2 documents interim corrective actions and four long-term stability options: Enhanced Option I-A, Option I-D, Option II, and Option III. A generic evaluation was performed for the interim corrective actions in Section 3.2.1 of Reference 2. This generic evaluation continues to be applicable for CPPU. Interim corrective action stability boundaries are kept the same in terms of absolute core power and flow; power levels, reported as a percentage of rated power, are scaled based on the new uprated power. For the long-term options, evaluations are core reload dependent and are performed for each reload fuel cycle. The analyses of each long-term option are addressed below. The topics addressed in this evaluation are:

Topic	CPPU Effect	Disposition
2.4.1 Enhanced Option I-A	May change stability regions and associated trip setpoint	[
2.4.2 Option I-D	May change exclusion region and SLMCPR protection may be affected	
2.4.3 Option II	May change exclusion region and SLMCPR protection may be affected	
2.4.4 Option III (OPRM armed region and trip setpoint)	OPRM Armed Region rescaled with power. Trip setpoint may change	

Topic	CPPU Effect	Disposition
2.4.4 Option III (Hot channel oscillation magnitude)	None	]

2.4.1 Plants with Enhanced Option I-A

CPPU Effect: The stability regions and associated trip setpoints may change with CPPU.

CPPU Basis: Enhanced Option I-A (E1A) is a prevention solution. Plants with the E1A stability solution have analytically based flow biased APRM flux trip functions (Exclusion and Restricted Regions) and an administratively controlled Monitored Region that are expressed as a percent of rated power. CPPU will not affect the Period Based Detection System feature of this stability option because it is power independent. The flow-biased APRM flux scram and control rod block trip functions, and the Monitored Region are subject to two effects. First, these features, when expressed in terms of percent power, must be rescaled to preserve the absolute power value. Second, they have a weak dependency on reload core design. As a result, these features are either confirmed or adjusted for each plant reload. [ ]the trip function settings and Monitored Region for power uprate will be established by the [ ] analysis that incorporates the new rated power level. [ ]

2.4.2 Plants with Option I-D

CPPU Effect: The exclusion region may change and SLMCPR protection may be affected by CPPU.

CPPU Basis: 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 detect-and-suppress feature is a demonstration that regional mode reactor instability is not probable and the existing flow-biased flux trip provides adequate SLMCPR protection for events, which initiate along the rated rod line.

Similar to the discussion in Section 2.4.1, 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 will also affect 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. [ ]

2.4.3 Plants with Option II

CPPU Effect: The exclusion region may change and SLMCPR protection may be affected by CPPU.

CPPU Basis: Option II is a detect-and-suppress solution, which applies to the two BWR/2 plants designed with a quadrant based APRM trip system. This quadrant-based system will detect either core-wide or regional mode instability. Plants implementing Option II must demonstrate that the flow-biased APRM flux trip is adequate to provide protection for the SLMCPR for events that initiate along the rated rod line.

Option II plants may also include an administratively controlled exclusion region. However, this exclusion region is dependent upon the fuel design and core loading and is reviewed and adjusted, as required, for each reload core. Because of this dependency, these features will be analyzed for the first reload analysis that incorporates the new rated power level.

CPPU 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 analysis. [ ]these features will be analyzed for the [ ] new rated power level. [ ]

2.4.4 Plants with Option III

CPPU Effect: The Option III trip setpoint may be affected by CPPU operating conditions. The OPRM Armed Region will be rescaled with CPPU.

CPPU Basis: Option III is a detect-and-suppress solution, which combines closely spaced LPRM detectors into “cells” to effectively detect any mode of reactor instability. Plants implementing Option III must demonstrate that the Option III trip setpoint is adequate to provide SLMCPR protection for anticipated reactor instability. This evaluation is dependent upon the core and fuel design and is performed for each reload. [ ]

The Option III automatic scram is provided by the Oscillation Power Range Monitor (OPRM). The generic analyses for the Option III hot channel oscillation magnitude and the OPRM hardware were designed to be independent of core power. [

]

The Option III trip is armed only when plant operation is within the Option III trip-enabled region. The Option III trip-enabled region is defined as the region on the power/flow map with power ≥30% OLTP and core flow ≤60% rated core flow. For CPPU, the Option III trip-enabled region is rescaled to maintain the same absolute power/flow region boundaries. Because the rated core flow is not changed, the 60% core flow boundary is not rescaled. The 30% OLTP boundary changes by the following equation:

EPU Region Boundary = 30% OLTP * (100% ÷ EPU (% OLTP))

Thus, for a 120% OLTP EPU:

EPU Region boundary = 30% OLTP * (100% ÷ 120%) = 25% EPU

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 power uprate. The effect on the nuclear characteristics of the fuel is discussed in Section 2.3. The topics considered in this section are:

Topic	CPPU Effect	Disposition
2.5.1 Scram Time Response (BWR/6)	Increase transient pressure response	[
2.5.1 Scram Time Response (BWR/2-5)	None	
2.5.2 CRD Positioning	Slight increase in pressure above core plate	
2.5.2 CRD Cooling	Slight increase in pressure above core plate	
2.5.3 CRD Integrity	Increased transient pressure response	]

2.5.1 Control Rod Scram

CPPU Effect: For pre-BWR/6 product lines, the scram times are decreased by the transient pressure response, [ ] For BWR/6 plants, the increase in the transient pressure response due to CPPU increases the scram time.

CPPU Basis: For pre-BWR/6 plants at normal operating conditions, the accumulator supplies the initial scram pressure and, as the scram continues, the reactor becomes the primary source of pressure to complete the scram. [

]

For BWR/6 plants at normal operating conditions, the accumulator supplies all of the pressure to complete the scram. Because the normal reactor dome pressure for CPPU does not change, the scram time performance relative to current plant operation is essentially the same. Therefore, BWR/6 plants will retain their current technical specification scram requirements.

[ ] The overpressure evaluation described in Section 3.1 will be used[ ] the transient scram times and accumulator pressure will be re-evaluated to account for CPPU effects. The revised scram times will then be used in the plant specific reload analysis core design, if necessary.

2.5.2 Control Rod Drive Positioning and Cooling

CPPU Effect: The increase in reactor power at the CPPU operating condition results in [ ] from the CRD System to the CRDs during normal plant operation.

CPPU Basis: [ ] and the automatic operation of the system flow control valve maintains the required drive water pressure and cooling water flow rate. [ ] The CRD cooling and normal CRD positioning functions are operational considerations and not safety related functions.

2.5.3 Control Rod Drive Integrity Assessment

CPPU Effect: [ ] The transient pressures due to uprated power may create higher pressure loadings.

CPPU Basis: 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 [ ] dispositioned in Section 3.3.2.

3.0 REACTOR COOLANT AND CONNECTED SYSTEMS

This section addresses the evaluations in Regulatory Guide 1.70, Chapter 5 and part of Chapter 3, that are documented in the current plant submittals. These reactor coolant and connected systems evaluations include:

Section	Title	Generic	Plant Specific
3.1	Nuclear System Pressure Relief/Overpressure Protection	[	
3.2	Reactor Vessel		
3.3	Reactor Internals		
3.4	Flow-Induced Vibration		
3.5	Piping Evaluation		
3.6	Reactor Recirculation System		
3.7	Main Steamline Flow Restrictors		
3.8	Main Steamline Isolation Valves		
3.9	Reactor Core Isolation Cooling		
3.10	Residual Heat Removal System		
3.11	Reactor Water Cleanup System		]

The detailed assessment dispositions as outlined in Section 1.1 are provided in the applicable sections. The plant specific evaluations will be reported in the plant specific submittal consistent with the format and level of detail of previous extended power uprate submittals or as indicated below. The applicability of the generic assessments for a specific plant application will be evaluated. The plant specific submittal will either document the successful confirmation of the generic assessment or provide a plant specific evaluation if the applicability assessment is unsuccessful.

3.1 NUCLEAR SYSTEM PRESSURE RELIEF/OVERPRESSURE PROTECTION

The nuclear system pressure relief system evaluation for the topics address in this evaluation is as follows:

Topic	CPPU Effect	Disposition
Overpressure capacity	Higher transient steam flow	[
Flow-induced vibration	Higher steam flow	]

CPPU Effect: For CPPU, the system operating pressure does not change but the steam flow rate increases as shown in Table 1-3. The increased steam flow rate associated with uprated power may increase steam line vibration. The increased core steam generation also causes an increase in the pressurization during some transient events.

CPPU Basis: 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 relief, SRV, and/or spring safety valves, as applicable, along with other functions provide this protection. [ ] An evaluation will be 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 [ ] These evaluations will also confirm the [ ] If changes in the SRV setpoints are required for CPPU, the effect of the setpoint changes on the affected power uprate evaluations will be evaluated.

[ ]

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

Increased main steam line flow may affect flow-induced vibration of the piping and safety/relief valves during normal operation. The flow-induced vibration of the piping will be addressed by vibration testing during initial plant operation at the higher steam flow rates (Section 10.4). The vibration frequency, extent and magnitude depend upon plant specific parameters, valve locations, the valve design and piping support arrangements. Flow-induced vibration may increase incidents of valve leakage. However, plants currently have procedures to address leaking relief valves, SRV and safety valves. Flow-induced vibration 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 to utilities and is also addressed in their procedures. The consequences of a stuck open SRV have been previously considered in the plant specific safety analyses[

]

3.2 REACTOR VESSEL

The Reactor Pressure Vessel (RPV) structure and support components form a pressure boundary to contain 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 considered in this section are:

Topic	CPPU Effect	Disposition
3.2.1 Fracture Toughness	Increased fluence	[
3.2.2 Reactor Vessel Structural Evaluation (Components not significantly affected)	No increased flow, temperature, RIPDs, and other mechanical loads Or Fatigue usage ≤ 0.5	
3.2.2 Reactor Vessel Structural Evaluation (Affected components)	Increased flow, temperature, RIPDs, and other mechanical loads And Fatigue usage > 0.5	]

3.2.1 Fracture Toughness

CPPU Effect: CPPU may result in a higher operating neutron flux at the vessel wall, consequently increasing the integrated flux over time (fluence).

CPPU Basis: Plant specific vessel wall fluence analyses will be performed consistent with NRC-approved methods.

An increase in fluence will result in an increase in the RPV adjusted reference temperature (ART) and a decrease in upper shelf energy (USE). In the case where the beltline P/T curves are limiting, an increase in ART will also require a revision to the pressure-temperature (P/T) curves. If the fluence increases, then the increase in the ART and decrease in USE are evaluated according to Regulatory Guide 1.99, Revision 2 (Reference 4).

The USE at end of life must remain greater than the 50 ft-lb criterion of 10CFR50 Appendix G. If the material does not meet the 50 ft-lb criterion or if the available data is insufficient to determine the USE, an equivalent margin analysis (EMA) can be performed in accordance with 10CFR50 Appendix G. GE performed a generic evaluation to demonstrate equivalent margins for BWR material USE (Reference 5). The NRC approved the GE generic EMA evaluation by an NRC SER (Reference 6). A plant specific evaluation is required to demonstrate that the materials meet the limits required for the EMA.

If the P/T curves are beltline limited and the ART increases, then new P/T curves will be required. NRC 10CFR50 Appendix G specifies fracture toughness requirements to provide adequate margins of safety during operation. Appendix G of Section XI of the ASME Code (Reference 8) forms part of the basis for the requirements of 10CFR50 Appendix G. A change to the P/T curves will require a change to the Technical Specification.

3.2.2 Reactor Vessel Structural Evaluation

CPPU Effect: For most RPV components, the flow, temperature, RIPDs and other mechanical loads do not increase. Consequently, there is no change in stress or fatigue for these components. The feedwater (FW) nozzle, however, experiences an increase in the FW flow and temperature, resulting in an increase in stress and fatigue.

CPPU Basis: Reactor Vessel components are required to comply with the structural requirements of the ASME Boiler and Pressure Vessel Code applicable to the components at the time of construction or the governing code used in the stress analysis for a modified component.

Certain reactor vessel components may be generically dispositioned without detailed structural analysis. [

]

[ ]
any component with an increase in loads and with a fatigue usage greater than [ ] will also require a plant specific evaluation. The plant specific evaluation will be performed consistent with the methods documented in Appendix I of Reference 1.

3.3 REACTOR INTERNALS

The reactor internals include core support structure (CSS) and non-core support structure (non-CSS) components. The topics considered in this section are:

Topic	CPPU Effect	Disposition
3.3.1 Reactor Internals Pressure Differences	Increased pressure differences	[
3.3.2 Reactor Internals Structural Evaluation	Increased flow, temperature, RIPDs and various loads	
3.3.3 Steam Dryer Separator Performance	Increased steam moisture content and separator carryunder	]

3.3.1 Reactor Internal Pressure Difference

CPPU Effect: Higher pressure differences across internals due to higher core exit steam flow.

CPPU Basis: The increase in core average power results in higher core loads and reactor internal pressure differences due to higher core exit steam flow. The reactor internal pressure differences (RIPDs) are calculated for normal steady-state operation, Upset, Emergency and Faulted conditions, as applicable, consistent with the existing design basis. The process used for calculating the RIPDs is described in Section 5.5.1.1 of Reference 1. Minor components such as [
]

3.3.2 Reactor Internals Structural Evaluation

CPPU Effect: The typical loads considered in the power uprate structural evaluation of the internals include: dead weight, RIPDs, seismic loads, hydrodynamic containment loads, annulus pressurization loads, jet reaction loads, thermal load effects, flow loads, acoustic and flow-induced loads due to recirculation line break, and fuel lift loads, as applicable, consistent with the design basis. [
]

CPPU Basis: The power uprate assessment of the internals is performed for the Normal, Upset, Emergency and Faulted conditions, as applicable, consistent with the existing design basis. In cases where permanent structural modifications or permanent repairs have been performed to the internals, the modified configuration and the corresponding documentation will form the design basis, in conjunction with the original design basis, as applicable.

The structural integrity evaluation for the CPPU typically will include the following key internal components:

• Shroud	• Control rod drive mechanism
• Shroud support	• Shroud head and separators
• Core plate	• Access hole cover

• Top guide	• Jet pumps
• Fuel channel	• FW sparger
• Orificed fuel support	• Core spray line and sparger
• Control rod guide tube	• Steam dryer
• Control rod drive housing	• LPCI Coupling

Detailed evaluations of these components are not necessary if it is shown that the existing design basis loads bound the loads based on CPPU. Otherwise, a reconciliation of the load increase will be performed to confirm that the stresses remain within their acceptable limits consistent with existing design basis. The evaluation of irradiation-assisted stress corrosion cracking (IASCC) and flow-induced vibration (FIV) are covered in Sections 10.7 and 3.4.2, respectively.

The power uprate evaluation will also include the assessment of any existing flaw evaluations of documented degraded conditions (e.g., crack indications) that were not repaired as well as those that were repaired.

3.3.3 Steam Dryer/Separator Performance

CPPU Effect: The power uprate increases the steam flow from the core and to the main steam lines during normal plant operation. The increase in steam flow and pressure drops in the steam separators and dryer can affect the operational margins associated with normal reactor water level. The increased steam flow can also affect the moisture content of the steam and the carryunder flow returning to the downcomer.

CPPU Basis: The expected performance of the steam separators and dryer will be evaluated to determine if the steam leaving the reactor pressure vessel meets the turbine operational criteria.

3.4 FLOW-INDUCED VIBRATION

The flow-induced vibration (FIV) evaluation addresses the influence of an increase in flow during CPPU on reactor coolant pressure boundary (RCPB) piping, RCPB piping components and RPV internals. The topics considered in this section are:

Topic	CPPU Effect	Disposition
3.4.1 Structural Evaluation of Recirc Piping	No significant increase in recirc flow rate	[
3.4.1 Structural Evaluation of Main Steam and Feedwater Piping	Increased main steam and feedwater flow rates	
3.4.1 Safety Related Thermowells and Probes	Increased main steam and feedwater flow rates	

3.4.2 Structural Evaluation of core flow dependent RPV Internals	No increase in core flow	
3.4.2 Structural Evaluation of other RPV Internals	Increased feedwater, steam, and recirculation pump drive flow	]

3.4.1 FIV Influence on Piping

CPPU Effect: Main steam and feedwater flow rates increase.

CPPU Basis: The key RCPB piping systems addressed by this task are the Main Steam Line (MSL) Supply System, the Feedwater (FW) Piping System and the Reactor Recirculation Coolant (RRC) System within the containment. Key applicable structures include the MSL Supply System piping and suspension, the FW System piping and suspension, and the RRC System piping and suspension. In addition, branch lines attached to the MSL System piping or FW System piping are also considered. Recirculation flow will not be significantly increased during CPPU operation, which may require only a small increase in the RRC System flow rates. [

]

The MSL System piping and the FW System piping will have higher mass flow rates and flow velocities. The vibration levels of the MSL and the FW system piping are expected to increase [ ] Hence, a startup vibration test using remote vibration sensors, such as accelerometers or strain gages, mounted on representative portions of the MSLs and FW piping located inside the containment will be required during the initial implementation of CPPU.

In addition, the large bore MSL and FW piping outside of containment, that is accessible by plant personnel, should also be monitored by performing visual observations and by taking vibration measurements using hand-held vibration instruments during a walkdown of this piping. This walkdown should be performed during initial plant operation at the CPPU conditions. Areas outside of containment, that are inaccessible to plant personnel when the plant is at high power levels, may also require the installation of remote vibration monitoring sensors.

The safety related thermowells and probes in the MSL and FW piping systems will also be evaluated[ ].

3.4.2 FIV Influence on Reactor Internal Components

CPPU Effect: Increase in reactor coolant quality. Increase in feedwater, steam and recirculation pump drive flow.

CPPU Basis: The FIV evaluation addresses the influence of CPPU on components in the lower plenum, core region and other reactor vessel regions. The reactor internal components FIV of all instrumented BWRs were reviewed to assess their vibration level and effect due to power uprate. The vibration levels of components in the lower plenum (such as control rod guide tubes, in-core guide tubes and shroud support legs), as well as the components in the core region (such as fuel channels)[

]

Components in other regions that are affected by FIV due to the increase in feedwater, recirculation drive and steam flow will be evaluated on a plant specific basis. Components such as jet pump assemblies, jet pump sensing lines, feedwater sparger and steam separators are evaluated at the uprate power, maximum core flow point [

] The evaluation includes assessment of plant startup data, dynamic structural analysis and, if necessary, fatigue usage determination.

3.5 PIPING EVALUATION

3.5.1 Reactor Coolant Pressure Boundary Piping

The Reactor Coolant Pressure Boundary Piping systems evaluation consists of a number of safety related piping subsystems that move fluid through the reactor and other safety systems. The topics considered in this section are:

Topic	CPPU Effect	Disposition
Structural evaluation for unaffected safety related piping	No increased flow, pressure, temperature, and other mechanical loads	[
Structural evaluation for affected safety related piping	Increased flow, pressure, temperature, or other mechanical loads	]

CPPU Effect: For most piping systems, the flow, pressure, temperature, and mechanical loads will not increase. Consequently, there will be no change in stress and fatigue evaluations. The FW and MSL piping and associated branch piping up to the first anchor or support, however, will experience an increase in the flow, pressure and/or temperature, resulting in an increase in stress and fatigue.

CPPU Basis: The piping systems are required to comply with the structural requirements of the ASME Boiler and Pressure Vessel Code (or an equivalent Code) applicable at the time of construction or the governing code used in the stress analysis for a modified component.

The FW and MSL and associated branch piping systems require a plant specific evaluation. For other safety related piping systems with [

] any safety related system or portions of a system with an increase in flow, pressure, temperature or mechanical load will be evaluated on a plant specific basis consistent with the methods described in Appendix K of Reference 1.

For CPPU, there is no significant change in the maximum operating pressure, temperature and flow rate for the recirculation piping system and attached RHR piping system. [

] The same conclusion is also applicable for the Control Rod Drive (CRD) System.

[

]

Some segments of the safety related systems are [

]

Analysis of safety related thermowells should be evaluated [ ].

3.5.2 Balance-of-Plant Piping

The Balance-of-Plant Piping systems evaluation consists of a number of piping subsystems that move fluid through systems outside the Reactor Pressure Coolant Boundary Piping. The topics considered in this section are:

Topic	CPPU Effect	Disposition
Structural evaluation for unaffected non-safety related piping	No increased flow, pressure, temperature, and other mechanical loads	[
Structural evaluation for affected non-safety related piping	Increased flow, pressure, temperature, or other mechanical loads	]

CPPU Effect: For some piping systems, the flow, pressure, temperature, and mechanical loads will not increase. Consequently, there will be no change in stress and fatigue evaluations. The FW and MSL piping including the associated branch piping, however, will experience an increase in the flow, pressure and/or temperature, resulting in an increase in stress and fatigue.

CPPU Basis: Large bore and small bore ASME Class 1, 2, and 3, and B31.1 Code piping and supports not addressed in Section 3.5.1 and affected by CPPU will be evaluated for acceptability at the CPPU conditions. The evaluation of the BOP piping and supports will be performed in a manner similar to the evaluation of RCPB piping systems and supports (Section 3.5.1), using applicable ASME Section III, Subsections NB/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 will be used.

For some of the BOP piping, the loads and temperatures used in the analyses are dependent upon the containment hydrodynamic loads and short/long term temperature evaluation results (Section 4.1). Bounding hydrodynamic loads and short/long term torus/suppression pool temperatures due to a design basis loss-of-coolant accident (LOCA) were defined for current licensed power. [

]

[

]

3.6 REACTOR RECIRCULATION SYSTEM

Topic	CPPU Effect	Disposition
System evaluation	Increased drive flow	[
NPSH	Slight change	
Flow mismatch	Setpoint adjustment only	
Single loop operation	Setpoint adjustment only	]

CPPU Effect: Increased voids in the core during normal uprated power operation requires a slight increase in the recirculation drive flow to achieve the same core flow.

CPPU Basis: The generic assessment for the recirculation system documented in Section 4.5 of Reference 2, Supplement 1, Volume I remains valid for CPPU. The cavitation interlock remains the same in terms of absolute flow rates. Scaling changes to the power-flow map are administrative and do not affect safety related considerations.

[

]

The recirculation pump mismatch Technical Specification limits do not change as a percent of speed (for M-G plants) or flow (for FCV plants), but the percent power level at which the allowed mismatch limits are lowered is [

] The flow mismatch limits must be reviewed only if a detailed ECCS evaluation is required (Section 4.3).

The absolute power limit for single-loop operation (SLO) stays the same, requiring a proportional reduction in the percent of rated power at the uprate power level.

3.7 MAIN STEAMLINE FLOW RESTRICTORS

Topic	CPPU Effect	Disposition
Structural integrity	Increased steam flow	[ ]

CPPU Effect: At uprated power, the flow restrictors are required to pass a higher flow rate, which will result in an increased pressure drop.

CPPU Basis: 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. [

]

3.8 MAIN STEAM ISOLATION VALVES (MSIV)

Topic	CPPU Effect	Disposition
Isolation performance	Increased steam flow	[
Valve pressure drop	Increased steam flow	]

CPPU Effect: The power uprate increases the steam flow through the MSIVs and results in a higher pressure drop and can affect the isolation performance.

CPPU Basis: Adequacy of the MSIV isolation performance will be [
]. The increase in MSIV pressure drop will be evaluated[
].

3.9 REACTOR CORE ISOLATION COOLING/ISOLATION CONDENSER

The Reactor Core Isolation Cooling (RCIC) System provides inventory makeup to the reactor vessel when the vessel is isolated from the normal high pressure makeup systems. The Isolation Condenser (IC) removes decay heat from the reactor vessel while maintaining the vessel liquid inventory when the vessel is isolated from the normal heat sink and high pressure makeup systems. The topics addressed in this evaluation are:

Topic	CPPU Effect	Disposition
System performance and hardware (RCIC & IC)	None	[
Net positive suction head (RCIC)	None	]
Adequate core cooling for limiting LOFW events (RCIC)	Higher decay heat	Addressed in Section 9.1.3
Inventory makeup (RCIC) Operational Level 1 avoidance	Higher decay heat	Addressed in Section 9.1.3
Heat removal capability (IC)	Higher decay heat	Addressed in Section 9.1.3

CPPU Effect: The higher decay heat changes the response of reactor water level following a loss of feedwater event in which high pressure core injection (HPCI) or high pressure core spray (HPCS) is assumed to fail. There is no change to the normal reactor operating pressure or the SRV setpoints.

CPPU Basis: The RCIC System, utilized in all BWR/4, 5 and 6 and some BWR/3 plants, 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 coolant flow from the Feedwater System. The system design injection rate must be sufficient for compliance with the system limiting criteria 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. Thus, the adequacy of the RCIC injection rate to meet this design basis event is evaluated for the CPPU.

An operational requirement is that the RCIC System can restore the reactor water level while avoiding Automatic Depressurization System (ADS) timer initiation and MSIV closure activation functions associated with the low-low-low reactor water level setpoint (Level 1). This requirement is intended to avoid unnecessary initiations of safety systems. Many plants have elected to elevate the nominal ECCS/ADS initiation level setpoint to compensate for indicated instrument level errors resulting from drywell heating effects during a LOCA. Compliance with this operational objective is not achieved for these plants under pre-CPPU conditions and will not be achieved under CPPU conditions. Operator action to inhibit ADS actuation following transient events will preclude reactor depressurization, thus allowing the RCIC System to perform its design basis function. This requirement is not a safety related function and may be evaluated on a plant specific basis (see Section 9.1.3).

For a CPPU, there is no change to the normal reactor operating pressure and the SRV 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. 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 Condensate Storage Tank volume as the source of water. Therefore, the specified operational temperature limit for the process water does not change with power uprate. [

]

The Isolation Condenser (IC) System, utilized in BWR/2 and some BWR/3 plants, provides the equivalent function as the RCIC for isolation events and must satisfy the same requirements. The IC System removes decay heat from the vessel by condensing the steam generated by the decay heat and returning the condensate to the vessel. For a CPPU, there is no change to the normal reactor operating pressure and the SRV setpoints remain the same. There is no change to the maximum specified reactor pressure for IC System operation. Therefore, no changes are required to the system hardware. CPPU conditions may increase the amount of reactor inventory lost through the SRVs before the rate of steam generated in the vessel decreases to the capacity of the IC. The IC System startup delay may need to be reduced in order to limit the inventory loss to acceptable levels.

The reactor system response to a loss of feedwater transient with RCIC or IC is discussed in Section 9.1.3.

3.10 RESIDUAL HEAT REMOVAL SYSTEM

Topic	CPPU Effect	Disposition
LPCI mode	Higher decay heat	Addressed in Section 4.2.4
Suppression pool and containment spray cooling modes	Higher decay heat	Addressed in Section 4.1
Shutdown cooling mode	Higher decay heat	[
Steam condensing mode	Higher decay heat	]
Fuel pool cooling assist	Higher decay heat	Addressed in Section 6.3.1

CPPU Effect: The Residual Heat Removal (RHR) System is designed to restore and maintain the reactor coolant inventory following a LOCA and remove reactor decay heat following reactor shutdown for both normal, transient, and accident conditions. CPPU effect on the RHR System is caused by the higher decay heat in the core corresponding to the uprated power and the increased amount of reactor heat discharged into the containment during a LOCA.

CPPU Basis: The RHR System operates in various modes, depending on plant operating modes and as assumed in accident analyses.

The Low Pressure Coolant Injection (LPCI) mode, as it relates to the LOCA response, is discussed in Section 4.2.4.

The Suppression Pool Cooling (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 Containment Spray Cooling (CSC) mode reduces drywell pressure, drywell temperature, and suppression chamber pressure following an accident. The adequacy of these operating modes will be demonstrated by the containment analysis (Section 4.1).

The Shutdown Cooling (SDC) mode is designed to remove the sensible and decay heat from the reactor primary system during a normal reactor shutdown. This non-safety operational mode allows the reactor to be cooled down within a certain time objective, so that the SDC mode of operation will not become a critical path during refueling operations. Because the power uprate increases the reactor decay heat, a longer time is required for the reactor cool down. [

] A longer SDC time does not have an effect on plant safety[

]

The Steam Condensing (SC) mode is designed to maintain the reactor at a hot shutdown condition during reactor isolation without depressurizing while the equipment failure creating the isolation can be repaired. The SC mode is not safety related and is only an operational aid. It has been disabled at many BWRs. The CPPU increases the reactor decay heat, resulting in a higher demand on the SC mode. If there is a need for initiating the SPC mode due to the high suppression pool temperatures, the higher demand on the remaining RHR heat exchangers due to CPPU results in a short delay in establishing the SPC mode. This is a system operational objective and does not have an effect on plant safety.

The Fuel Pool Cooling 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. The adequacy of fuel pool cooling, including use of the Fuel Pool Cooling Assist mode, is addressed in Section 6.3.1.

3.11 REACTOR WATER CLEANUP SYSTEM

Topic	CPPU Effect	Disposition
System performance	Increased contaminant input rate	[ ]
Containment isolation	Increased feedwater line pressure	Addressed in Section 4.1

CPPU Effect: The Reactor Water Cleanup (RWCU) System may be slightly affected by the increase in feedwater flow due to the power uprate.

CPPU Basis: The RWCU System is a normally operating system with no safety related functions other than containment isolation. [

] The effect of this increase is included in Section 4.1 containment isolation assessment.

4.0 ENGINEERED SAFETY FEATURES

This section addresses the evaluations in Regulatory Guide 1.70, Chapter 6, that are documented in the current plant power uprate submittals. These engineered safety feature evaluations include:

Section	Title	Generic	Plant Specific
4.1	Containment System Performance	[	
4.2	Emergency Core Cooling Systems		
4.3	Emergency Core Cooling Systems Performance		
4.4	Main Control Room Atmosphere Control System		
4.5	Standby Gas Treatment System		
4.6	Main Steam Isolation Valve Leakage Control System		
4.7	Post-LOCA Combustible Gas Control		]

The detailed assessment dispositions as outlined in Section 1.1 are provided in the applicable sections. The plant specific evaluations will be reported in the plant specific submittal consistent with the format and level of detail of previous extended power uprate submittals or as indicated below. The applicability of the generic assessments for a specific plant application will be evaluated. The plant specific submittal will either document the successful confirmation of the generic assessment or provide a plant specific evaluation if the applicability assessment is unsuccessful.

4.1 CONTAINMENT SYSTEM PERFORMANCE

This section addresses the effect of the uprated power on various aspects of the containment system performance. The topics considered in this section are:

Topic	CPPU Effect	Disposition
Pool temperature response	Increase decay heat	[
Wetwell pressure	Increase decay heat	
Drywell temperature	Increase decay heat	
Drywell pressure	Increase decay heat	
Containment dynamic loads	Increase decay heat	
Containment isolation	Increase decay heat	
Motor-operated valves	Increase decay heat	

Topic	CPPU Effect	Disposition
Hardened wetwell vent system	Increase decay heat	
Equipment operability	Increase decay heat	]

CPPU Effect: The suppression pool temperature increases as a result of the higher decay heat associated with CPPU. The assumption of constant pressure minimizes the effect on other aspects of the containment evaluation.

CPPU Basis: The effect of CPPU on the containment system will be analyzed for each plant using the methods documented in Section 5.10.2 of Reference 1 and results will be reported in the plant specific power uprate submittal. This assessment will include the following:

- Pressure and temperature response
- Containment dynamic loads
- Containment isolation
- Compliance with Generic Letter 89-10
- Compliance with Generic Letter 89-16
- Compliance with Generic Letter 96-06
- Compliance with Generic Letter 95-07

The containment isolation evaluation will include manual-operated valves that experience a change in containment pressure on one side of the valve. Motor-operated valves are addressed in compliance with Generic Letter 89-10. Assessment of the hardened wetwell vent system is addressed in response to Generic Letter 89-16. Assessment of other equipment operability is addressed in compliance with Generic Letter 96-06. [
]

4.2 EMERGENCY CORE COOLING SYSTEMS

The emergency core cooling systems (ECCS) include the high pressure system (either High Pressure Coolant Injection (HPCI) or High Pressure Core Spray (HPCS)), the Core Spray (CS) or Low Pressure Core Spray (LPCS) system, the Low Pressure Coolant Injection (LPCI) mode of the RHR System, and the ADS. [
] The following topics are addressed:

Topic	CPPU Effect	Disposition
4.2.1 High Pressure Coolant Injection	None	[
4.2.2 High Pressure Core Spray	None	
4.2.3 Core Spray or Low Pressure Core Spray	None	

Topic	CPPU Effect	Disposition
4.2.4 Low Pressure Coolant Injection System	None	
4.2.5 Automatic Depressurization	None	
4.2.6 ECCS Net Positive Suction Head	Potential change in suppression pool temperature and containment pressure	]

4.2.1 High Pressure Coolant Injection

CPPU Effect: The increase in decay heat changes the response of the reactor water level following a small break LOCA or a loss of feedwater transient event. There is no change to the normal reactor operating pressure or the SRV setpoints.

CPPU Basis: The HPCI System, utilized in all BWR/4 and some BWR/3 plants, is designed to pump water into the reactor vessel over a wide range of operating pressures. The primary purpose of the HPCI 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. Although for this analysis, the HPCI System is typically assumed to be out of service, the adequacy of the HPCI System is demonstrated by the margins discussed in Section 4.3.

In addition, the HPCI System serves as a backup to the RCIC System to provide makeup water in the event of a loss of feedwater flow transient that is evaluated as described in Section 9.1. Because the HPCI injection rate is significantly greater than the RCIC injection rate, the adequacy of the HPCI System to meet the safety requirement following a loss of feedwater flow event is demonstrated by the discussion in Section 9.1.3.

For a CPPU, there is no change to the normal reactor operating pressure and the SRV setpoints remain the same. [

]

The NPSH available for the HPCI pump [

] The NPSH required by the HPCI pump [

]

4.2.2 High Pressure Core Spray

CPPU Effect: There is no change to the normal reactor operating pressure or the SRV setpoints.

CPPU Basis: The HPCS System utilized in BWR/5 and 6 plants is designed to spray water into the reactor vessel over a wide range of operating pressures and was evaluated in Section 4.3 of Reference 2. The HPCS System provides reactor vessel coolant inventory makeup in the event of a small break LOCA that does not immediately depressurize the reactor vessel and helps to depressurize the reactor vessel. This system also provides spray cooling for long-term core cooling after a LOCA. The adequacy of this system is demonstrated by the margins discussed in Section 4.3 and the containment evaluation (Section 4.1).

The HPCS System also serves as a backup to the RCIC System to provide makeup water in the event of a loss of feedwater flow transient, as described in Section 9.1.3. Because the HPCS injection results in RPV depressurization and the injection rate is significantly greater than the RCIC injection rate, the adequacy of the HPCS System to meet the safety requirement following a loss of feedwater flow event is demonstrated by the discussion in Section 9.1.3.

There is no change to the maximum specified reactor pressure for HPCS System operation [ ] The maximum injection pressure for the HPCS System is conservatively based on the upper analytical limit for the lowest available group of safety/relief valves. Because the SRV settings remain the same for the CPPU, [

]

4.2.3 Core Spray or Low Pressure Core Spray

CPPU Effect: There is no change in the reactor pressures at which the CS/LPCS is required.

CPPU Basis: The CS/LPCS System sprays water into the reactor vessel after it is depressurized. The primary purpose of the CS/LPCS 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 spray cooling for long-term core cooling in the event of a LOCA. The adequacy of the CS/LPCS System performance is demonstrated by the margins discussed in Section 4.3.

[

]

4.2.4 Low Pressure Coolant Injection

CPPU Effect: There is no change in the reactor pressures at which the LPCI mode of RHR is required.

CPPU Basis: The LPCI mode of the RHR System is automatically initiated in the event of a LOCA. The primary purpose of the LPCI System 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 power uprate. The adequacy of this system is demonstrated by the margins discussed in Section 4.3.

[
]

4.2.5 Automatic Depressurization System

CPPU Effect: The CPPU does not change the conditions at which the ADS must function.

CPPU Basis: The ADS uses relief or safety/relief valves 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/LPCS 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

CPPU Effect: CPPU rated thermal power operation increases the reactor decay heat, which increases the heat addition to the suppression pool following a LOCA, Station Blackout and Appendix R events. As a result, the long-term peak suppression pool water temperature and long-term peak containment pressure increase. The most limiting case for NPSH typically occurs at the peak long-term suppression pool temperature.

CPPU Basis: The ECCS NPSH was evaluated in Section 4.1.8.5 of Reference 2, Supplement 1, Volume I. Similarly, for HPCI, HPCS, CS/LPCS and RHR/LPCI Systems, changes in the peak long-term suppression pool temperature and containment pressure are determined by the containment analyses (Section 4.1). If these values are bounded by the previous evaluation, no additional plant specific analyses are required for the NPSH.

4.3 EMERGENCY CORE COOLING SYSTEM PERFORMANCE

The 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 will be performed to demonstrate that the 10CFR50.46 requirements continue to be met at the uprated power conditions. The topics addressed in this evaluation are:

Topic	CPPU Effect	Disposition
Large break peak clad temperature – limiting case	Small effect	[
Large break peak clad temperature – limiting event analysis	Small effect	
Small break peak clad temperature – break spectrum	Increased decay heat	
Small break peak clad temperature – ADS capacity	Increased decay heat	
Local cladding oxidation	Negligible effect	
Core wide metal water reaction	No effect	
Coolable geometry	No effect	
Long-term cooling	No effect	]

CPPU Effect: CPPU has only a small effect on the limiting large break LOCA Peak Cladding Temperature (PCT). The small break LOCA PCT may increase due to the higher decay heat.

CPPU Basis: [

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

]

4.4 MAIN CONTROL ROOM ATMOSPHERE CONTROL SYSTEM

Topic	CPPU Effect	Disposition
Iodine intake	Increased source term	[ ]

CPPU Effect: CPPU increases the radioisotopes seen by the control room atmosphere control system following an accident.

CPPU Basis: The effect on the control room intake filters due to the increased iodine release rate will be evaluated. The filter loading will be calculated for the limiting design basis accident, considering the differences in timing of releases and atmospheric dispersion coefficients for each accident.

4.5 STANDBY GAS TREATMENT SYSTEM

Topic	CPPU Effect	Disposition
Flow capacity	None	[
Iodine removal capability	Increased iodine loading	]

CPPU Effect: The core inventory of iodine and subsequent loading on the Standby Gas Treatment System (SGTS) filters or charcoal adsorbers are affected by CPPU.

CPPU Basis: The 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 design basis accident.

The design flow capacity of the system 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 typically present so that the post-LOCA iodine loading on the charcoal remains below the guidance provided by Regulatory Guide 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, [ ] analyses have been performed to evaluate 1) SGTS at facilities that have received approval under 10 CFR 50.67 to implement an Alternate Source Term and 2) SGTS at facilities committed to Regulatory Guide 1.3 for fission product transport. [

]
Based on the parameter values above, the AST evaluation results in a maximum charcoal loading of [ ] of total iodine per gram of charcoal, [ ]the 2.5 mg/gm maximum value in Regulatory Guide 1.52. In addition, the decay heating analysis based on the above values results in a maximum component temperature of [ ] with normal flow conditions and [ ] under conditions of a failed fan with minimum cooling flow. These temperatures are within the [ ] SGTS component temperature limit and the [ ] charcoal ignition temperature.

The Regulatory Guide 1.3 decay heating analysis, based on the above parameter values, results in a maximum component temperature of [ ] with normal flow conditions and [ ] under conditions of a failed fan with minimum cooling flow. Again, these temperatures are within the [ ] SGTS component temperature limit and the [ ] charcoal ignition temperature, and are valid for both the Regulatory Guide 1.52 adsorber

iodine loading of 2.5 mg/gm as well as for the 60 mg/gm loading for facilities not committed to the adsorber bed-sizing criterion of Regulatory Guide 1.52.
The individual plant values of the bounding parameters will be confirmed enveloped for plant specific applications.

4.6 MAIN STEAM ISOLATION VALVE LEAKAGE CONTROL SYSTEM

Topic	CPPU Effect	Disposition
Radiological effect	Increased source term	[ ]

CPPU Effect: The radioisotopes released through the MSIVs during an accident will increase due to the CPPU.

CPPU Basis: Most BWR plants do not have a MSIV Leakage Control system. The plants that have this system send the leakage flow back to the reactor or to the secondary containment where the radiation is handled by the Standby Gas Treatment System. CPPU will not significantly affect the leakage flow rate. The increase in radiological sources has no effect for systems that send the leakage back to the reactor. For those systems that send the leakage to the secondary containment, the SGTS handles the additional radioisotopes as described in Section 4.5.

4.7 POST-LOCA COMBUSTIBLE GAS CONTROL SYSTEM

Topic	CPPU Effect	Disposition
System initiation time	Increased radiolysis	[
Recombiner operating temperature	Increased radiolysis	
Nitrogen makeup	Increased radiolysis	]

CPPU Effect: As a result of CPPU, the post-LOCA production of hydrogen and oxygen by radiolysis increases [ ].

CPPU Basis: 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. Because of the increased production of hydrogen and oxygen, an earlier start of the system occurs when the procedurally controlled limits have been reached.

The need for an earlier start of the system after the accident does not affect the ability of operators to respond to the event, because the system is typically started hours or days following the event. [

] For hydrogen mixing systems used in Mark III containment designs, initiation time is either (1) automatic based on LOCA signals such as containment pressure, (2) procedurally controlled based on a set time after the LOCA event, or (3) based on containment hydrogen concentration. [

]

[

]

[

]

5.0 INSTRUMENTATION AND CONTROL

This section addresses the evaluations in Regulatory Guide 1.70, Chapter 7, that are documented in the current plant power uprate submittals. The principal instrumentation and control evaluations and summary disposition of these evaluations are as follows:

Section	Title	Generic	Plant Specific
5.1	NSSS Monitoring and Control	[	
5.2	BOP Monitoring and Control		
5.3	Technical Specification Instrument Setpoints		]

The detailed assessment dispositions as outlined in Section 1.1 are provided in the applicable Sections. The plant specific evaluations will be reported in the plant specific submittal consistent with the format and level of detail of previous extended power uprate submittals or as indicated below. The applicability of the generic assessments for a specific plant application will be evaluated. The plant specific submittal will either document the successful confirmation of the generic assessment or provide a plant specific evaluation if the applicability assessment is unsuccessful.

5.1 NSSS MONITORING AND CONTROL

The instruments that monitor and the controls that 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 those instrument allowable values and/or setpoints for those NSSS sensed variables, which initiate protective actions. The effect of CPPU on Technical Specifications is addressed in Section 5.3. The topics considered in this section are:

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

5.1.1 Neutron Monitoring System

CPPU affects the performance of the Neutron Monitoring System. The specific performance effects are associated with the Average Power Range Monitors (APRMs), Intermediate Range Monitors (IRMs), Local Power Range Monitors (LPRMs), Rod Block Monitor (RBM), and Rod Worth Minimizer (RWM) or Rod Control Information System (RCIS). The following evaluations of the Neutron Monitoring System are applicable to GE or Reuter Stokes supplied monitoring equipment or vendor monitoring equipment that meets the GE design specifications at power uprate conditions.

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

CPPU Effect: At rated power, the increase in power level increases the average flux in the core and at the in-core detectors.

CPPU Basis: The APRM power signals are calibrated to read 100% at the new licensed power. 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.

5.1.1.2 Local Power Range Monitors

CPPU Effect: At rated power, the increase in power level increases the flux at the LPRMs.

CPPU Basis: Due to the increase in neutron flux experienced by the LPRMs and traversing incore probes (TIPs), it is expected that the neutronic life of the LPRM detectors will 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. TIPs are stored in shielded rooms. A small increase in radiation levels can be accommodated by the radiation protection program for normal plant operation.

5.1.1.3 Rod Block Monitor

CPPU Effect: 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.

CPPU Basis: 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.

5.1.2 Rod Worth Minimizer/Rod Control and Information System

CPPU Effect: The increase in power level could change the power level at which rod patterns are enforced by the RWM or RCIS.

CPPU Basis: The RWM and RCIS are normal operating systems that do not perform a safety related function. The function of the RWM and RCIS Rod Pattern Controller is to support the operator by enforcing rod patterns until reactor power has reached appropriate levels. The RCIS also provides rod position information to the operator. The RCIS Rod Withdrawal Limiter prevents excessive control rod withdrawal after reactor power has reached an appropriate level. Therefore, no additional plant specific information for the performance of these systems relative to the normal operational function is required. The power-dependent instrument setpoints for both the RWM and RCIS rod pattern controller (BWR/6) are included in the plant Technical Specifications (Section 5.5.3).

5.2 BOP MONITORING AND CONTROL

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

Topic	CPPU Effect	Disposition
5.2.1 Pressure Control System	Increased power level and steam flow	[
5.2.2 Turbine Steam Bypass System (Normal Operation)	Increased power level and steam flow	
5.2.2 Turbine Steam Bypass System (Safety Analysis)	Increased power level and steam flow	
5.2.3 Feedwater Control System (Normal Operation)	Increased power level and feedwater flow	
5.2.3 Feedwater Control System (Safety Analysis)	Increased power level and feedwater flow	
5.2.4 Leak Detection System	Increased feedwater temperature	]

5.2.1 Pressure Control System

CPPU Effect: The increase in power level increases the steam flow to the turbine.

CPPU Basis: The Pressure Control System (PCS) is a normal operating system that provides fast and stable responses to system disturbances related to steam pressure and flow changes so that reactor pressure is controlled within its normal operating range. As noted in Reference 1, Appendix F, 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 that adequate turbine control valve pressure control and flow margin is available.

5.2.2 Turbine Steam Bypass System

CPPU Effect: The bypass system capacity in terms of mass flow is not changed for CPPU. As a result, the increase in power level and resulting increase in steam flow to the turbine effectively reduces the bypass system in terms of percent of uprated steam flow.

CPPU Basis: 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 anticipated operational occurrence (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

CPPU Effect: The increase in power results in an increase in feedwater flow.

CPPU Basis: The Feedwater Control System is a normal operation system that controls the water supply to the reactor to maintain water level. 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. A loss of feedwater event can be caused by downscale failure of the controls. The loss of feedwater flow is discussed in Section 9.1.3.

5.2.4 Leak Detection System

CPPU Effect: The only effect on the Leak Detection System due to CPPU is a slight increase in the feedwater temperature and steam flow.

CPPU Basis: [

] The

increased feedwater temperature results in a small increase in the main steam tunnel temperature. [

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

5.3 TECHNICAL SPECIFICATION INSTRUMENT SETPOINTS

Technical Specifications instrument allowable values and/or setpoints are those sensed variables, which initiate protective actions and are generally associated with the safety analysis. Technical Specification allowable values are highly dependent on the results of the safety analysis. The safety analysis generally establishes the analytical limits. The determination of the Technical Specification allowable values and other instrument setpoints includes consideration of measurement uncertainties and is derived from the analytical limits. The settings are selected with sufficient margin to minimize inadvertent initiation of the protective action, while assuring that adequate operating margin is maintained between the system settings and the actual limits. There is typically substantial margin in the safety analysis process that should be considered in establishing the setpoint process used to establish the Technical Specification allowable values and other setpoints.

Increases in the core thermal power and steam flow affect some instrument setpoints. These setpoints are adjusted to maintain comparable differences between system settings and actual limits, and reviewed to ensure that adequate operational flexibility and necessary safety functions are maintained at the uprated power level. Where the power increase results in new instruments being employed, an appropriate setpoint calculation is performed and Technical Specification changes are implemented, as required. [

]

[

] The justification for implementing this simplified process for the individual Technical

Specification setpoints is provided for each instrument below. Implementing the constant maximum operating pressure requirement for CPPU [
]

In addition, the following restrictions are imposed on the use of the simplified process to assure its validity. Its use is limited to:

- NRC approved GE or plant specific methodology.
  [
  ]

The topics considered in this section are:

Topic	CPPU Effect	Disposition
5.3.1 Main Steam Line High Flow Isolation - Setpoint Calculation Methodology	Increased reactor power level and steam flow	[
5.3.1 Main Steam Line High Flow Isolation - Setpoint Value	Increased reactor power level and steam flow	
5.3.2 Turbine First-Stage Pressure Scram Bypass - Setpoint Calculation Methodology	Increased reactor power level and turbine first-stage pressure change	
5.3.2 Turbine First-Stage Pressure Scram Bypass - Setpoint Value	Increased reactor power level and turbine first-stage pressure change	
5.3.3 APRM Flow-Biased Scram - Setpoint Calculation Methodology	Increased reactor power level	
5.3.3 APRM Flow-Biased Scram – Setpoint Value	Increased reactor power level	
5.3.4 Rod Worth Minimizer/ RCIS Rod Pattern Controller Low Power Setpoint - Setpoint Calculation Methodology	Increased reactor power level, turbine first stage pressure change, and increased feedwater flow	
5.3.4 Rod Worth Minimizer/ RCIS Rod Pattern Controller Low Power Setpoint – Setpoint Value	Increased reactor power level, turbine first stage pressure change, and increased feedwater flow	
5.3.5 Rod Block Monitor	Increased reactor power level	

Topic	CPPU Effect	Disposition
5.3.6 RCIS Rod Withdrawal Limiter High Power Setpoint - Setpoint Calculation Methodology	Increased reactor power level and turbine first stage pressure change	
5.3.6 RCIS Rod Withdrawal Limiter High Power Setpoint - Setpoint Value	Increased reactor power level and turbine first stage pressure change	
5.3.7 APRM Setdown in Startup Mode - Setpoint Calculation Methodology	Reduced safety limit for reduced pressure or low core flow conditions	
5.3.7 APRM Setdown in Startup Mode - Setpoint Value	Reduced safety limit for reduced pressure or low core flow conditions	]

5.3.1 Main Steam Line High Flow Isolation

CPPU Effect: Increased reactor power level and steam flow.

CPPU Basis: This setpoint is used to isolate the Group 1 primary containment isolation valves. The only safety analysis event that credits this trip is the main steamline break accident. For this accident, there are diverse trips from high area temperature and high area differential temperature. The analytical limit for high main steamline flow isolation for CPPU is maintained at the current percent (e.g., 140%) of rated steam flow in each main steam line as long as the main steam line flow rate limiter choked flow capability is not exceeded. However, the MSL flow rate is monitored using differential pressure and the psid value is included in some plant Technical Specifications. [

] The main steamline flow restrictor limits coolant lost through the break and the subsequent radioactive exposure. However, the radiological analysis is based on the capability of the flow restrictor, which limits the break to typically between 170% and 200% of normal steam flow at original licensed thermal power. The main steamline high flow analytical limit is typically 140% of normal steam flow at uprated thermal power conditions, which is well below the analysis value. [

] Also, the Technical Specification limit may be reduced in some cases to ensure that this limit is below the main steam line flow restrictor capability.

A Technical Specification change may be required (1) if a new instrument is required to monitor the increased differential pressure; (2) to assure that the Technical Specification limit is below the main steamline flow restrictor choke flow capability; or (3) to change the differential pressure at the allowable steam flow.

5.3.2 Turbine First-Stage Pressure Scram and Recirculation Pump Trip Bypass

CPPU Effect: Increased reactor power level and potential change to turbine first-stage pressure.

CPPU Basis: The turbine first-stage pressure setpoint is used to reduce scrams and recirculation pump trips at low power levels where the turbine steam bypass system is effective for turbine trips and generator load rejections. In the safety analysis, this trip bypass only applies to events at low power levels that result in a turbine trip or load rejection. [

]

[

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

5.3.3 APRM Flow-Biased Scram

CPPU Effect: Increased reactor power level.

CPPU Basis: This scram is not specifically credited in any safety analysis event. [

]

The Technical Specifications will be modified by adjusting the flow-biased scram setpoint.

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

CPPU Effect: Increased reactor power level, potential change to turbine first-stage pressure, and increased feedwater flow.

CPPU Basis: The Rod Worth Minimizer/RCIS Rod Pattern Controller Low Power Setpoint is used to bypass the rod pattern constraints established for the control rod drop accident at low power levels. The consequences of the CRDA are acceptable above 10% CLTP, and the rod pattern constraints are no longer necessary. The sensing point for this instrument is generally either the feedwater flow or turbine first-stage pressure. [

]

[

] To ensure that the new value is appropriate, power uprate plant ascension startup test or normal plant surveillance is used to validate that the actual plant interlock is cleared consistent with the safety analysis. A Technical Specification change may be required if the instruments are rescaled or the HPT modified.

5.3.5 Rod Block Monitor

CPPU Effect: Increased reactor power level.

CPPU Basis: The severity of rod withdrawal error during power operation event is dependent upon the RBM rod block setpoint. This setpoint is only applicable to the control rod withdrawal error. [

]

5.3.6 RCIS Rod Withdrawal Limiter High Power Setpoint

CPPU Effect: Increased reactor power level.

CPPU Basis: The RCIS RWL is a BWR6 system that is only used in the analysis of the control rod withdrawal error analysis**.** [

]

[

] To ensure that the new value is appropriate, power uprate plant ascension startup test or normal plant surveillance is used to validate that the actual plant interlock is cleared consistent with the safety analysis. A Technical Specifications change may be required if the HPT is modified.

5.3.7 APRM Setdown in Startup Mode

CPPU Effect: Reduced Technical Specification safety limit for reduced pressure or low core flow conditions

CPPU Basis: The APRM setdown in the startup mode provides margin to the safety limit. Further, critical power tests demonstrated that the safety limit is conservative. A diverse trip is provided by the IRMs**.** The value for the Technical Specification safety limit for reduced pressure or low core flow conditions may be reduced to satisfy the fuel thermal monitoring requirements established as described in Section 2.1. The APRM setdown in the startup mode setpoint is based on the Technical Specification setpoint.

The current Technical Specification may be based on either a conservative generic setpoint or on a plant specific calculated value.

[

]

6.0 ELECTRICAL POWER AND AUXILIARY SYSTEMS

This section addresses the evaluations in Regulatory Guide 1.70, Chapters 8 and 9, that are documented in the current plant power uprate submittals. The principal electrical power and auxiliary systems evaluations and summary disposition of these evaluations are as follows:

Section	Title	Generic	Plant Specific
6.1	AC Power	[	
6.2	DC Power		
6.3	Fuel Pool		
6.4	Water Systems		
6.5	Standby Liquid Control		
6.6	Power Dependent HVAC		
6.7	Fire Protection		
6.8	Other Systems Affected by Power Uprate		]

The detailed assessment dispositions as outlined in Section 1.1 are provided in the applicable sections. The plant specific evaluations will be reported in the plant specific submittal consistent with the format and level of detail of previous extended power uprate submittals or as indicated below. The applicability of the generic assessments for a specific plant application will be evaluated. The plant specific submittal will either document the successful confirmation of the generic assessment or provide a plant specific evaluation if the applicability assessment is unsuccessful.

6.1 AC POWER

The AC power supply includes both off-site and on-site power. The on-site power distribution system consists of transformers, buses, and switchgear. Alternating current (AC) power to the distribution system is provided from the transmission system or from onsite Diesel Generators. The topics considered in this section are:

Topic	CPPU Effect	Disposition
AC power (degraded voltage)	Increased power output and normal operating loads	[
AC power (normal operation)	Increased power output and normal operating loads	]

CPPU Effect: The increase in thermal power from the reactor translates to an increased electrical output from the station. The increased normal operating loads depend on the specific plant design and may include: the recirculation pumps, condensate pumps, condensate booster pumps, motor driven feedwater pumps, and circulating water pumps. The safety related electrical loads are not significantly increased.

CPPU Basis: For the off-site power supply, the equipment is typically adequate for operation with the uprated electrical output. Changes in electrical requirements to support normal plant operation are not safety related. The increased power from the generator may have some effect on the grid stability/reliability. A grid stability analysis will be performed, and the results of the analysis summarized in the plant specific submittal. Any plant changes to control the reactive power will be identified in the plant specific submittal. The protective relaying for the main generator may require changing. Any changes will be identified in the plant specific submittal.

Station loads under emergency operation/distribution conditions (emergency diesel generators) are based on equipment nameplate data, except for the ECCS pumps where a conservatively high flow brake horsepower (BHP) is used. Operation at the uprated level is achieved by utilizing existing equipment operating at or below the nameplate rating and within the calculated BHP for the stated pumps; therefore, under emergency conditions, the electrical supply and distribution components are considered adequate.

[

] evaluation of the AC power system is performed to assure an adequate AC power supply to safety related systems.

6.2 DC POWER

The direct current (DC) power distribution system provides control and motive power for various systems/components within the plant. The topics considered in this section are:

Topic	CPPU Effect	Disposition
DC power requirements	No significant effect	[ ]

CPPU Effect: There is no significant effect.

CPPU Basis: [

] System loads are computed based on equipment nameplate data. Operation at the uprated level is not expected to increase any loads beyond nameplate rating or revise any control logic. [ ] the DC power system is performed to assure an adequate DC power supply to safety related systems.

6.3 FUEL POOL

The following topics are addressed in this section:

Topic	CPPU Effect	Disposition
6.3.1 Fuel Pool Cooling (normal core offload)	Increased heat load	[
6.3.1 Fuel Pool Cooling (full core offload)	Increased heat load	
6.3.2 Crud Activity and Corrosion Products	Increase source term	
6.3.3 Radiation Levels	Increase source term	
6.3.4 Fuel Racks	Increased heat load	]

6.3.1 Fuel Pool Cooling

CPPU Effect: For the same time after shutdown, the spent fuel pool heat load increases due to the decay heat generation as a result of the power uprate.

CPPU Basis: The spent fuel pool temperature must be maintained below the licensing limit (140 to 150°F). The limiting condition is typically a full core discharge with all remaining spaces filled with used fuel from prior discharges. A normal offload is typically considered in outage planning with the additional assumption of a redundant train out of service. In some cases, the RHR Fuel Pool Assist mode may be used to augment the capacity of fuel pool cooling. The temperature requirement assures operator comfort and provides ample margin against an inventory loss in the fuel pool due to evaporation or boiling, which is an operational requirement.

If there are difficulties in meeting the temperature limit, the start of transfer of the spent fuel to the spent fuel pool after reactor shutdown can be delayed to reduce the heat load to an acceptable level. In addition, the total bundle transfer duration to the spent fuel pool can be increased.

6.3.2 Crud Activity and Corrosion Products

CPPU Effect: Crud activity and corrosion products associated with spent fuel can increase slightly due to power uprate.

CPPU Basis: The amount of crud activity and pool quality are operational considerations and are unrelated to safety. [
] and fuel pool water quality is maintained by the Fuel Pool Cleanup System.

6.3.3 Radiation Levels

CPPU Effect: The normal radiation levels around the pool may increase slightly primarily during fuel handling operation.

CPPU Basis: The potential for increased occupational exposure is an operational consideration and unrelated to safety. [

]

6.3.4 Fuel Racks

CPPU Effect: The increased decay heat from the CPPU results in a higher heat load in the racks during long-term storage.

CPPU Basis: The fuel racks are designed for higher temperatures than the licensing limit. The Fuel Pool Cooling System assures that the licensing limit is maintained.

6.4 WATER SYSTEMS

The water systems are designed to provide a reliable supply of cooling water for normal operation and design basis accident conditions. The topics considered in this section are:

Topic	CPPU Effect	Disposition
Water systems performance (normal operation)	Increased heat loads	[
Water systems performance (safety related)	Increased heat load	
Suppression pool cooling (RHR service operation)	Increased decay heat rate	
Ultimate heat sink	Increased heat load	]

CPPU Effect: CPPU results in increased heat load during normal operation and in a greater decay heat rate, which increase the safety related water systems cooling requirements during accident conditions.

CPPU Basis: The performance of the safety related Service Water System during and immediately following the most limiting design basis event, the LOCA,[

]

The containment analysis (Section 4.1) will determine if additional RHR service water cooling capacity is required as a result of the increased decay heat rate. If CPPU does not increase the cooling requirements on the RHR System and its associated Service Water System, no changes are required. If additional cooling capacity is needed, a modification will be made to assure that adequate cooling is available.

The ultimate heat sink (UHS) temperature may be affected by the increase in normal operating heat load. For most plants, the environmental effects of uprate are controlled at the same level as is presently in place. That is, the plant operation is managed such that none of the present limits such as maximum allowed ultimate heat sink temperature is increased as a result of uprate. However, for some plants, there may be a small change in UHS temperature.

6.5 STANDBY LIQUID CONTROL SYSTEM

Topic	CPPU Effect	Disposition
Core shutdown margin	Fuel design dependent	[
System performance and hardware	Increased heat load and potential increase in transient reactor pressure	
Suppression pool temperature following limiting ATWS event	Increased core power	]

CPPU Effect: Changes in the fuel design for CPPU may require modifications to the Standby Liquid Control (SLC) System as the result of reductions in the reactor shutdown margin and increases in the suppression pool temperature for the limiting ATWS event.
[

]

CPPU Basis: The SLC System, utilized in all BWR plants, is designed to pump a neutron absorber solution into the reactor vessel over a wide range of reactor operating pressures. The SLC is designed to shut down the reactor from uprated power conditions to cold shutdown in the postulated situation that none of the control rods can be inserted. This is typically a manually operated system that pumps a sodium pentaborate solution into the vessel, to provide neutron absorption and achieve a subcritical reactor condition.

The power increase alone does not affect the requirements for the minimum reactor boron concentration. [

] An increase in the reactor boron concentration may be achieved by increasing, either individually or collectively, (1) the minimum solution volume, (2) the minimum specified solution concentration, or (3) the isotopic enrichment

of the Boron-10 in the stored neutron absorber solution. [
]

The SLCS is typically designed for injection at a maximum reactor pressure equal to the upper analytical setpoint for the lowest group of SRVs operating in the relief mode. [

](see Section 9.3.1).

The ATWS analysis for uprated power conditions (Section 9.3.1) may impose new boron injection rate requirements for the purpose of maintaining the peak suppression pool temperature within established limits during the limiting ATWS event. An increase in the reactor boron injection rate may be achieved by increasing, either individually or collectively, (1) the pump capacity, (2) the minimum specified solution concentration, or (3) the isotopic enrichment of the Boron-10 in the stored neutron absorber solution.

6.6 POWER DEPENDENT HEATING, VENTILATION AND AIR CONDITIONING

The Heating, Ventilation and Air Conditioning (HVAC) systems consist mainly of heating, cooling supply, exhaust and recirculation units in the turbine building, reactor building and the drywell, which support normal plant operation. The topics considered in this section are:

Topic	CPPU Effect	Disposition
Power dependent HVAC performance	Increased heat loads	[ ]

CPPU Effect: CPPU results in slightly higher process temperatures and electrical loads.

CPPU Basis: CPPU is expected to result in slightly higher process temperatures (e.g., feedwater temperature) and a small increase in the heat load due to higher electrical currents in some motors and cables. All of these are operational considerations.

[

]

6.7 FIRE PROTECTION

This section addresses the effect of CPPU on the fire protection program, fire suppression and detection systems, reactor and containment system responses to postulated 10CFR50 Appendix R fire events. The topics addressed in this evaluation are:

Topic	CPPU Effect	Disposition
Fire suppression and detection systems	None	[
Operator response time	Increased decay heat	
Peak cladding temperature	Increased decay heat	
Vessel water level	Increased decay heat	
Suppression pool temperature	Increased decay heat	]

CPPU Effect: The higher decay heat associated with CPPU may reduce the time available for the operator to perform the actions necessary to achieve and maintain cold shutdown conditions. The higher decay heat also results in higher suppression pool temperatures. The higher decay heat may result in lower vessel water levels or higher peak cladding temperatures (depending on the plant specific analysis basis).

CPPU Basis: [

] Therefore, the reactor and containment responses and operator actions will be evaluated [ ] for CPPU as described in Section 5.11.1 of Reference 1.

6.8 OTHER SYSTEMS AFFECTED BY POWER UPRATE

The topics considered in this section are:

Topic	CPPU Effect	Disposition
Other systems	None or not significant	[ ]

CPPU Effect: The CPPU does not have any significant effect on other systems not addressed in this report.

CPPU Basis: [

]

7.0 POWER CONVERSION SYSTEMS

This section addresses the evaluations in Regulatory Guide 1.70, Chapter 10, that are documented in the current plant power uprate submittals. These power conversion system evaluations include:

Section	Title	Generic	Plant Specific
7.1	Turbine-Generator	[	
7.2	Condenser and Steam Jet Air Ejectors		
7.3	Turbine Steam Bypass		
7.4	Feedwater and Condensate		]

The detailed assessment dispositions as outlined in Section 1.1 are provided in the applicable sections. The plant specific evaluations will be reported in the plant specific submittal consistent with the format and level of detail of previous extended power uprate submittals or as indicated below. The applicability of the generic assessments for a specific plant application will be evaluated. The plant specific submittal will either document the successful confirmation of the generic assessment or provide a plant specific evaluation if the applicability assessment is unsuccessful.

7.1 TURBINE-GENERATOR

The turbine-generator converts the thermal energy in the steam into electrical energy. The topics considered in this section are:

Topic	CPPU Effect	Disposition
Turbine-generator performance	Increased power level and steam flow	Plant Specific
Turbine-generator missile avoidance	Missile probability	[ ]

CPPU Effect: The increase in thermal energy and steam flow from the reactor is translated to an increased electrical output from the station by the turbine-generator. The increase in steam flow can also change the previous missile avoidance and protection analysis

CPPU Basis: The turbine-generator is required for normal plant operation and is not safety related.

Most plants were originally designed for a maximum of steam flow of 105%. Experience with previous power uprate applications indicates that turbine and generator modifications (e.g., turbine rotating element modification) are required to support power uprate. These modifications are required to support normal operation and are non-safety related.

The only safety related evaluation is the plant specific turbine-generator missile avoidance and protection analysis. The entrapped energy following a turbine trip or load rejection increases slightly for CPPU. Relative to the turbine generator missile protection analysis, many power plants have replaced high pressure and low pressure shrunk-on rotors with an integral rotor without shrunk-on wheels. These integral rotors are not considered a source for potential missile generation for CPPU for the slight increase in entrapped energy;[ ] An evaluation is required for rotors with shrunk-on wheels. The turbine generator overspeed protection systems will be evaluated to ensure that adequate protection is provided for CPPU conditions

7.2 CONDENSER AND STEAM JET AIR EJECTORS

The 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 (SJAE) remove noncondensable gases from the condenser to improve thermal performance. The topics considered in this section are:

Topic	CPPU Effect	Disposition
Condenser and SJAE	Increased power level and steam flow	[ ]

CPPU Effect: The increase in steam flow increases the heat removal requirement for the condenser. The additional power level increases the noncondensable gases generated by the reactor.

CPPU Basis: The condenser and SJAE functions are required for normal plant operation and are not safety related.

Most plants were originally designed with condensers and SJAEs that had extra capacity. [ ] These potential modifications support normal operation and are non-safety related.

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 considered in this section are:

Topic	CPPU Effect	Disposition
Turbine steam bypass (normal operation)	Increased power level and steam flow	[
Turbine steam bypass (safety analysis)	Increased power level and steam flow	]

CPPU Effect: The increase in steam flow reduces the relative capacity of the Turbine Steam Bypass System.

CPPU Basis: The Turbine Steam Bypass System is required for normal plant maneuvering and transients, and is not safety related.

[

] The Turbine Steam Bypass System is a normal operating system and non-safety related.

The actual 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 topics considered in this section are:

Topic	CPPU Effect	Disposition
Feedwater and condensate systems	Increased power level and feedwater flow	[ ]

CPPU Effect: The increase in power level increases the feedwater requirements of the reactor.

CPPU Basis: The Feedwater and Condensate Systems are required for normal plant operation and are not safety related.

[

] These modifications are required to support normal operation and are non-safety related.

8.0 RADWASTE AND RADIATION SOURCES

This section addresses the evaluations in Regulatory Guide 1.70, Chapter 11, that are documented in the current plant power uprate submittals. The radwaste and radiation source evaluations include:

Section	Title	Generic	Plant Specific
8.1	Liquid Waste Management	[	
8.2	Gaseous Waste Management		
8.3	Radiation Sources in the Reactor Core		
8.4	Radiation Sources in the Reactor Coolant		
8.5	Radiation Levels		
8.6	Normal Operation Off-Site Doses		]

The detailed assessment dispositions as outlined in Section 1.1 are provided in the applicable sections. The plant specific evaluations will be reported in the plant specific submittal consistent with the format and level of detail of previous extended power uprate submittals or as indicated below. The applicability of the generic assessments for a specific plant application will be evaluated. The plant specific submittal will either document the successful confirmation of the generic assessment or provide a plant specific evaluation if the applicability assessment is unsuccessful.

8.1 LIQUID AND SOLID WASTE MANAGEMENT

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

Topic	CPPU Effect	Disposition
Coolant fission and corrosion product levels	Slightly elevated levels	Addressed in Section 8.4
Waste Volumes	Slight increase	[ ]

CPPU Effect: Increased power levels and steam flow result in the generation of slightly higher levels of liquid and solid radwaste and coolant concentrations of fission and corrosion products.

CPPU Basis: Coolant activation and Corrosion products are slightly increased as a result of CPPU as discussed in Section 8.4.

The single largest source of liquid and wet solid waste is from the backwash of condensate demineralizers. CPPU results in an increased flow rate through the condensate demineralizers, resulting in a slight reduction in the average time between backwashes. This reduction does not affect plant safety. Similarly, the reactor water cleanup (RWCU) filter-demineralizer requires more frequent backwashes due to slightly higher levels of activation and fission products.

[

] are made to assess the operational impact of increased waste processing and to assure there are no significant environmental effects.

8.2 GASEOUS WASTE MANAGEMENT

Topic	CPPU Effect	Disposition
Offsite release rate	Small effect	[
Recombiner performance	Increased radiolysis	]

CPPU Effect: Under CPPU conditions, core radiolysis increases linearly with reactor thermal power, thus increasing the heat load on the offgas recombiner and related components. Other functions of the Offgas System are not significantly affected by power uprate.

CPPU Basis: 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 as low as reasonably achievable (ALARA) and does not exceed applicable guidelines. The radiological release rate is administratively controlled to remain within existing 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 [ ] relative to the design power level, without exceeding the design basis temperatures, flow rates, or heat loads.

[

]

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. Topics covered in this section are:

Topic	CPPU Effect	Disposition
Post operational radiation sources for radiological and shielding analysis	Radiation Sources increase proportional to power	[ ]

CPPU Effect: Core radiation sources increase proportional to the increase in reactor power.

CPPU Basis: 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 are needed for post-accident and spent fuel pool 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 Mega-Watt of reactor thermal power at various times after shutdown.

A bounding analysis has been performed to envelop the radiation sources evaluation for [

]

Individual plant values of these bounding parameters will be confirmed enveloped by the correspondent bounding values.

The results of this assessment will be used in performing analyses identified in Sections 8.5, 9.2, and 10.3.

8.4 RADIATION SOURCES IN REACTOR COOLANT

Radiation sources in the reactor coolant include activation products, activation corrosion products, and fission products. A [ ] assessment is provided for each of these sources and is divided into the following topics:

Topic	CPPU Effect	Disposition
8.4.1 Coolant Activation Products	Increased neutron flux increases production rate of activation products	[
8.4.2 Activated Corrosion Products and Fission Products	Increased neutron flux increases activation rate for corrosion products and increases fission products	]

8.4.1 Coolant Activation Products

CPPU Effect: Increases in reactor power will increase the activity of activation products found in reactor coolant.

CPPU Basis: During reactor operation, the coolant passing through the core region becomes radioactive as a result of nuclear reactions. The coolant activation is the dominant source in the turbine building and in the lower regions of the drywell. Because these sources are produced by interactions in the core region, their rates of production are proportional to power. The activation of the water is in approximate proportion to the increase in thermal power. [

] The typical margin in the plant design basis for reactor coolant concentrations significantly exceeds the potential increases due to power uprate, which will be verified by a plant specific evaluation. Because the transport time from core exit to downstream points will decrease with increased flow from CPPU, the resultant dose rates in the main steam lines, turbines, and condenser area will increase roughly proportional to power uprate and is determined by [ ].

8.4.2 Activated Corrosion Products and Fission Products

CPPU Effect: Increases in reactor power will increase the activity of corrosion products and fission products found in reactor coolant.

CPPU Basis: 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 CPPU conditions, the feedwater flow increases with power, the activation rate in the reactor region increases with power, and the filtration run-lengths of the condensate demineralizers may decrease as a result of the feedwater flow increase. The net result tends to increase 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. The noble gases released during plant operation result from the escape of minute fractions of the fission products in the fuel rods. An increase in this fractional release as a result of power uprate is not expected though the absolute rate of noble gases in the steam will increase roughly in proportion to the power uprate reflecting the increase in noble gas inventory in the fuel rods themselves. This escaped activity is the noble gas offgas that is included in the plant design. The original design basis was selected to be 0.1 curies/sec after 30 minutes decay. [

]

The fission product activity in the reactor water, like the activity in the steam, is the result of minute releases from the fuel rods. As is the case for the noble gases, [

]

Overall, the increase in fission product concentrations in reactor water and steam will result in higher levels in the water and steam [ ] Nevertheless, in specific areas where radionuclides may be concentrated, the resultant radiation fields may increase by more than the percentage increase in EPU [ ]

8.5 RADIATION LEVELS

Radiation levels during operation are derived from coolant sources and are covered under the following topic:

Topic	CPPU Effect	Disposition
Normal operational radiation levels	Radiation levels increase slightly	[
Post-operation radiation levels	Radiation levels increase slightly	
Post-accident radiation levels	Radiation levels increase slightly	]

CPPU Effect: For CPPU, normal operation radiation levels increase slightly.

CPPU Basis: [ ] [ ] In addition, plants employing Hydrogen Water Chemistry often exceed the original basis for shielding in the turbine building and offsite and are licensed under results of empirical analysis for operation with HWC. Such plants require specific reanalysis of radiation fields directly affected by HWC for changes in radiation zoning and compliance to 10 CFR 50, Appendix I and 40 CFR 190.

[ ] Regardless, individual worker exposures will be maintained within acceptable limits by the site ALARA program, which controls access to radiation areas. Procedural controls will compensate for increased radiation levels. In addition, plants with cobalt reduction, zinc injection, hydrogen water chemistry and/or noble metal chemical addition programs, are expected to show a decrease in post-operation radiation levels and/or reduced repairs required in radiation areas.

[ ]the impact of the increased core inventory on commitments made relevant to NUREG-0737 items (for which dose calculations were made) needs to be performed and the evaluation and its results described in the power uprate [ ] submittal. Examples include: post-accident sampling system, post-accident vital area access, post-accident effluent radiation monitors, and technical support center habitability.

[ ]assessment of normal operational radiation increases and radiation zoning or shielding in the various areas of the plant will be made and procedural controls will be generated to compensate for increased radiation levels.

8.6 NORMAL OPERATION OFF-SITE DOSES

The primary source of normal operation offsite doses is (1) airborne releases from the Offgas System and (2) gamma shine from the plant turbines. The following topics are considered:

Topic	CPPU Effect	Disposition
Plant gaseous emissions	Gaseous releases from offgas increase proportional to power	[
Plant skyshine from the turbine	Increase is directly proportional to the increase in rated steam flow	]

CPPU Effect: For CPPU, normal operation gaseous activity levels increase slightly, while the level of N-16 in the turbine increases in proportion to the rated steam flow.

CPPU Basis: The sources responsible for offsite dose increase by varying factors depending upon the basis for each source. [ ] The Technical Specifications limits implement the guidelines of 10CFR50, Appendix I. A review of the doses allowed by Technical Specifications limits is required to determine if sufficient margin is available to accommodate this increase. Power uprate does not involve significant increases in the offsite dose from noble gases, airborne particulates, iodine, tritium or liquid effluents. Present offsite radiation levels are a negligible portion of background radiation. [ ] below the limits of 10CFR20 and 10CFR50, Appendix I.

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. Typical shielding design more than adequately bounds increases due to power uprate. However, for plants that incorporate hydrogen water chemistry in addition to power uprate, a site-specific analysis will confirm the adequacy of shielding and protection for both plant personnel and the public.

[ ] assessment will be made of increases in airborne releases and corresponding offsite integrated doses and concentrations to confirm compliance with the limits of 10CFR20, 10CFR50, Appendix I, and 40CFR190. In addition, if hydrogen water chemistry is in use, an assessment of the increase in offsite skyshine will be made and confirmation of compliance to the limits of 10CFR50, Appendix I and 40 CFR 190 determined.

9.0 REACTOR SAFETY PERFORMANCE EVALUATIONS

This section addresses the evaluations in Regulatory Guide 1.70, Chapter 15, that are documented in the current plant power uprate submittals. These reactor safety performance evaluations include:

Section	Title	Generic	Plant Specific
9.1	Anticipated Operational Occurrences	[	
9.2	Design Basis Accidents		
9.3.1	ATWS		
9.3.2	Station Blackout		]

The detailed assessment dispositions as outlined in Section 1.1 are provided in the applicable sections. The plant specific evaluations will be reported in the plant specific submittal consistent with the format and level of detail of previous extended power uprate submittals or as indicated below. The applicability of the generic assessments for a specific plant application will be evaluated. The plant specific submittal will either document the successful confirmation of the generic assessment or provide a plant specific evaluation if the applicability assessment is unsuccessful.

9.1 ANTICIPATED OPERATIONAL OCCURRENCES

The Anticipated Operational Occurrence (AOO) events previously identified to be reviewed for an extended power uprate are given in Table E-1 of Reference 1. These AOO events include fuel thermal margin and loss of water level events. Also included in this table are two overpressure protection analysis events that are addressed in Section 3.1 of this report. The fuel thermal margin events are used to determine the fuel operating limit MCPR. Both the Thermal Margin and Loss of Water Level events are discussed below for CPPU. The topics addressed in this evaluation are:

Topic	CPPU Effect	Disposition
9.1.1 Fuel Thermal Margins Events	Small effect	[
9.1.2 Power and Flow Dependent Limits	Small effect	
9.1.3 Loss of Water Level Events (Loss of feedwater flow)	Increased decay heat	
9.1.3 Loss of Water Level Events (Loss of one feedwater pump)	Increased decay heat	]

9.1.1 Fuel Thermal Margin Events

CPPU Effect: Minor change in MCPR due to change in core hydraulics and flatter radial power profile and core hydraulics at CPPU power levels.

CPPU Basis: [

]

[

]

[

]

9.1.2 Power and Flow Dependent Limits

CPPU Effect: Not affected by CPPU.

CPPU Basis: The operating 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

CPPU Effect: Higher decay heat results in a lower reactor water level for loss of water level events.

CPPU Basis: [

]

For the Loss of Feedwater Flow 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 will be performed as described in Section 5.3.2 of Reference 1. This analysis will use the limiting high pressure or heat removal makeup system (IC, RCIC or HPCS). To be consistent with the accepted practice for the application of best-estimate decay heat models, the Loss of Feedwater Flow evaluation will use the ANSI/ANS 5.1-1979 decay heat standard with a two sigma uncertainty. As discussed in Section 3.9, there is an operational requirement for the RCIC System to restore the reactor water level, while avoiding the Automatic Depressurization System (ADS) timer initiation and MSIV closure activation functions associated with the low-low-low reactor water level setpoint (Level 1). This requirement is intended to avoid unnecessary initiations of safety systems. This requirement is not a safety related function and may not be evaluated on a plant specific basis.

Loss of One Feedwater Pump was included in Reference 1 only for operational considerations. As stated in the NRC Safety Evaluation, Section 4.5, to Reference 2, "A plant specific analysis of the loss of one feedwater pump event will be submitted per Appendix E of ELTR1 to assess the effect of a higher flow control line on scram avoidance". Since CPPU does not include an increase in the MELLLA upper boundary, the loss of one feedwater pump event is not significantly affected and therefore does not needed to be evaluated and included in the plant specific power uprate submittal.

9.2 DESIGN BASIS ACCIDENTS

This section addresses the radiological consequences of Design Basis Accident (DBA) analysis for existing analysis that complies with either TID-14844 based standards and regulations referred to as 10CFR100 or Regulatory Guide 1.183 regulations and standards referred to as 10CFR50.67.

Primary to the analysis of most design basis accidents is the inventory of fission product radionuclides in the reactor core since the core is the single largest source of radioactive materials in a nuclear power plant. For CPPU calculations, whether analyzed with respect to 10CFR100 or 10CFR50.67, a generic inventory listing of fission product radionuclides has been developed as is described in Section 8.3. [

]

9.2.1 10CFR100

This section concerns application to plants licensed under the requirements of 10CFR100 as interpreted based upon TID-14844 as a precedent. The topics addressed in this evaluation are:

Topic	CPPU Effect	Disposition
Main Steamline Break outside containment	None	[
Instrument Line Break	None	
LOCA inside containment	Increased source term	
Fuel Handling Accident	Increased source term	
Control Rod Drop Accident	Increased source term	
Other DBA analyzed in UFSAR	To Be Determined	]

CPPU Effect: The higher core power level increases the source term for the radiological release to the environment.

CPPU Basis: The magnitude of radiological consequences of a design basis accident (DBA) is basically 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 mechanisms between the core and the release point. For most DBAs, assuming no change in transport mechanisms, the radiological releases under CPPU are expected to increase proportional to the core inventory increase for offsite dose calculations. [

]

[

]

The result of such evaluations will provide whole body and thyroid dose at the exclusion area boundary and low population zone and skin dose in the main control room in addition to whole body and thyroid dose.

[

]

[

]

[

]

[

]

9.2.2 10CFR50.67

This section concerns application to plants licensed under the requirements of 10CFR 50.67 as interpreted based upon Regulatory Guide 1.181 and SRP 15.0.1. The topics addressed in this evaluation are:

Topic	CPPU Effect	Disposition
Design Basis Accidents	Increased source term	[ ]

CPPU Effect: The higher core power level increases the source term for the radiological release to the environment. Other factors affecting the transport analysis for fission product transport to the environment may also be affected by CPPU.

CPPU Basis: The magnitude of radiological consequences of a design basis accident (DBA) is basically 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 mechanisms between the core and the release point. Unlike analysis performed under prior TID-14844 rules, which were primarily prescriptive, analysis performed in accordance with Regulatory Guide 1.183 contain a significant amount of detail with respect to the mechanistic response of the plant to design basis accident conditions. [

]

The result of such evaluations will provide evaluations of dose commitment to members of the public in accordance with the requirements of 10CFR50.67 or 10CFR100 as applicable and dose evaluations for operators in accordance with 10CFR50, GDC 19.

9.3 SPECIAL EVENTS

This section considers three special events: Anticipated Transients without Scram (ATWS), Station Blackout (SBO), and ATWS with core instability.

Topic	CPPU Effect	Disposition
9.3.1 ATWS (Overpressure) – Event Selection	Higher power	[
9.3.1 ATWS (Overpressure) – Limiting Events	Higher power	
9.3.1 ATWS (Suppression Pool Temperature) - Event Selection	Higher steam discharge	
9.3.1 ATWS (Suppression Pool Temperature) – Limiting Events	Higher steam discharge	
9.3.1 ATWS (Peak Cladding Temperature)	Negligible effect	
9.3.2 Station Blackout	Increased decay heat	
9.3.3 ATWS with Core Instability	Core design	]

9.3.1 Anticipated Transients Without Scram

CPPU Effect: The higher operating steam flow will result in higher peak vessel pressures. The higher power and decay heat will result in higher suppression pool temperatures The increased core power and reactor steam flow rates, in conjunction with the SRV capacity and response times, could impact the capability of the SLCS to mitigate the consequences of an ATWS event.

CPPU Basis: [ ]ATWS evaluation is required for CPPU. This evaluation will be performed using the methodology documented in Section 5.3.4 of Reference 1 and will meet the following criteria:

- Maintain reactor vessel integrity (i.e., peak vessel bottom pressure less than the ASME service level C limit of 1500 psig).
- Maintain containment integrity (i.e., maximum containment pressure and temperature lower than the design pressure and temperature of the containment structure).
- Maintain coolable core geometry.

The evaluation will include consideration of the most limiting RPV overpressure and suppression pool temperature cases. Previous evaluations considered four ATWS events. [

]

Coolable core geometry is assured by meeting the 2200°F peak cladding temperature and the 17% local cladding oxidation acceptance criteria of 10CFR50.46. [

]

The evaluation will include consideration of the effect of RPV pressure response during the time the SLC System is required to inject into the reactor for mitigation of an ATWS event. Effects on the SLC System process parameters and design requirements will be determined and addressed as part of the system evaluation (Section 6.5).

9.3.2 Station Blackout

CPPU Effect: The plant responses to and coping capabilities for SBO event are affected slightly by operation at the power uprate level, due to the increase in the decay heat.

CPPU Basis: SBO will be reevaluated using the guidelines of NUMARC 87-00 and NRC Regulatory Guide 1.155, and consistent with the plant specific licensing basis. [

]

9.3.3 ATWS With Core Instability

CPPU Effect: 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 will not significantly affect the event progression.

CPPU Basis: The NRC has reviewed and accepted GE's disposition of the impact of large coupled thermal-hydraulic/neutronic core oscillations during a postulated ATWS event, presented in NEDO-32047-A, "ATWS Rule Issues Relative to BWR Core Thermal-Hydraulic Stability" (Reference 15). The companion report, NEDO-32164, "Mitigation of BWR Core Thermal-Hydraulic Instabilities in ATWS," (Reference 16) was approved by the same SER. The NRC review concluded that the GE TRACG code is an adequate tool to estimate the behavior of operating reactors during transients that may result in large power oscillations. The review also concluded that the severity of the event indicates that core coolable geometry and containment integrity can be maintained, and specified operator actions are sufficient to mitigate the consequences of an ATWS event with large core power oscillations.

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 NEDO-32047-A and NEDO-32164 was performed for an assumed plant initially operating at OLTP and MELLLA minimum flow point. [

]

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

]

Initial operating conditions of Feedwater Heater Out of Service (FWHOOS) and Final Feedwater Temperature Reduction (FFWTR) do not significantly impact the ATWS instability response reported in NEDO-32047-A and NEDO-32164. The limiting ATWS evaluation assumes that all feedwater heating is lost during the event and the injected feedwater temperature approaches the lowest achievable main condenser hot well temperature. [

]

10.0 OTHER EVALUATIONS

This section addresses the evaluations in Section 10 and specific plant unique items from Section 11 of the current plant extended power uprate submittals. The major evaluations and summary disposition of these evaluations are as follows:

Section	Title	Generic	Plant Specific
10.1	High Energy Line Break	[	
10.2	Moderate Energy Line Break		
10.3	Environmental Qualification		
10.4	Testing		
10.5	Individual Plant Evaluation		
10.6	Operator Training and Human Factors		
10.7	Plant Life		
10.8	NRC and Industry Communications		
10.9	Emergency Operating Procedures		]

The detailed assessment dispositions as outlined in Section 1.1 are provided in the applicable sections. The plant specific evaluations will be reported in the plant specific submittal consistent with the format and level of detail of previous extended power uprate submittals or as indicated below. The applicability of the generic assessments for a specific plant application will be evaluated. The plant specific submittal will either document the successful confirmation of the generic assessment or provide a plant specific evaluation if the applicability assessment is unsuccessful.

10.1 HIGH ENERGY LINE BREAK

High energy line breaks (HELBs) are evaluated for their effects on equipment qualification. The topics addressed in this evaluation are:

Topic	CPPU Effect	Disposition
Steam lines	No effect	[
Liquid lines	Increased subcooling	]

CPPU Effect: No effect on steam line breaks because steam conditions at the postulated break locations are unchanged. CPPU may increase subcooling in the reactor vessel, which may lead to increased break flow rates for liquid line breaks.

CPPU Basis: [

]

CPPU conditions may result in an increase in the mass and energy release for liquid line breaks. Therefore, liquid line breaks will be evaluated for CPPU. The evaluations will include EPU effects on subcompartment pressures and temperatures, pipe whip and jet impingement and flooding, consistent with the plant licensing basis.

10.2 MODERATE ENERGY LINE BREAK

Moderate energy line breaks (MELBs) are evaluated for their effects on equipment qualification. The topics addressed in this evaluation are:

Topic	CPPU Effect	Disposition
Flooding	No effect	[
Environmental Qualification	Increase in fluid temperature	]

CPPU Effect: CPPU results in no change in the inventory contained in moderate energy lines. The fluid process temperatures may increase which may lead to an increase in subcompartment atmospheric temperatures.

CPPU Basis: [

] the moderate energy liquid line break effect on environmental qualification will be evaluated [ ] for CPPU. The effect on environmental qualification is addressed in Section 10.3.

10.3 ENVIRONMENTAL QUALIFICATION

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

Topic	CPPU Effect	Disposition
10.3.1 Electrical Equipment	Power and radiation levels increase	[
10.3.2 Mechanical Equipment With Non-Metallic Components	Power and radiation levels increase	
10.3.3 Mechanical Component Design Qualification	Power and radiation levels increase	]

10.3.1 Electrical Equipment

CPPU Effect: The increase in power level increases the radiation levels experienced by equipment during normal operation and accident conditions. [

]

CPPU Basis: The safety related electrical equipment is reviewed for CPPU to ensure the existing qualification for the normal and accident conditions expected in the area where the devices are located remain adequate. Conservatisms in accordance with IEEE 323 are applied to the environmental parameters as required.

Environmental qualification (EQ) for safety related electrical equipment located inside the containment is based on main steam line break 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 [ ]will be evaluated through the EQ temperature monitoring program, which tracks such information for equipment aging considerations. [

] The plant environmental envelope for radiation is reviewed to determine if the current envelope is exceeded. If it is exceeded, the qualification of the equipment located within the containment will be reviewed.

Accident temperature, pressure, and humidity environments used for qualification of equipment outside containment result from a main steam line break in the steam tunnel, or other high energy line breaks, whichever is limiting for each plant area. [

] Maximum accident radiation levels used for qualification of equipment outside containment are from a DBA/LOCA. The plant environmental envelope for radiation is reviewed to determine if the current envelope is exceeded. If it is exceeded, the qualification of the equipment located within the containment will be reviewed.

10.3.2 Mechanical Equipment With Non-Metallic Components

CPPU Effect: The increase in power level increases the radiation levels experienced by equipment during normal operation and accident conditions. [

]

CPPU Basis: [

] The accident radiation level and the normal radiation level also increase slightly due to uprate. The equipment with non-metallic components is evaluated as discussed in Section 10.3.1.

10.3.3 Mechanical Component Design Qualification

CPPU Effect: The increase in power level increases the radiation levels experienced by equipment during normal operation and accident conditions. [

]

CPPU Basis: The mechanical design of equipment/components (e.g., heat exchangers) in certain systems is affected by operation at the uprate power level due to slightly increased flow and, in some cases, temperatures. [

]

The effects of increased fluid-induced loads on safety related components are described in Section 3 and evaluated in the containment loads analysis (Section 4.1). Increased nozzle loads and component support loads due to the uprated operating conditions are evaluated within the piping assessments in Section 3.4. [

] the adequacy of the mechanical component design qualification will be provided in the plant specific submittal.

10.4 TESTING

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

Topic	CPPU Effect	Disposition
Testing	Power level increase	[ ]

CPPU Effect: The increase in power level changes plant and system performance.

CPPU Basis: Based on the analyses and experience with uprated plants, a standard set of tests have 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-EPU rated thermal power, so that system performance parameters can be projected for uprate power before the pre-EPU power rating is exceeded.
- CPPU power increases will be made along an established flow control/rod line in increments of ≤ 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, pressure controls, and recirculation flow controls, if applicable. 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 and recirculation pump trip 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. [

]

[

]

Further, the important nuclear characteristics required for transient analysis are confirmed by the steady state physics testing. Transient mitigation capability is demonstrated by other tests required by the Technical Specifications. In addition, the limiting transient analyses are included as part of the reload licensing analysis.

10.5 INDIVIDUAL PLANT EVALUATION

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

Topic	CPPU Effect	Disposition
10.5.1 Initiating Event Frequency	Power level increase	[

Topic	CPPU Effect	Disposition
10.5.2 Component Reliability	Power level increase	
10.5.3 Operator Response	Power level increase	
10.5.4 Success Criteria	Power level increase	
10.5.5 External Events	Power level increase	
10.5.6 Shutdown Risk	Power level increase	
10.5.7 PRA Quality	No direct effect	]

Sections 10.5.1 through 10.5.4 address the CPPU effect on internal events PRA, Section 10.5.4 on external events PRA and Section 10.5.6 on shutdown risk. Section 10.5.7 addresses the quality requirements for the PRA.

The effect of CPPU on plant risk, including core damage frequency (CDF) and Large Early Release Fraction (LERF) will be evaluated on a plant specific basis. Factors to be considered in this assessment are discussed below. The effect of CPPU on the PRA will be provided in the plant specific submittal, including a description and quantification of the effect of CPPU on CDF and LERF.

10.5.1 Initiating Event Frequency

CPPU Effect: The increase in power level results in the plant operating closer to limits, which can potentially increase event frequency and affect CDF and LERF results.

CPPU Basis: [

] The plant specific submittal will identify and address the risk acceptability of any equipment that exceeds its operating limits, conditions, and/or ratings. The CPPU effects will be determined when the plant specific PRA is revised and a description of each of these effects, as well as their quantified impacts on CDF and LERF, will be provided in the plant specific submittal.

10.5.2 Component and System Reliability

CPPU Effect: The minimum acceptable required system or component capability may increase as a result of the increased power level, which may affect the system or component reliability and CDF and LERF results.

CPPU Basis: [

] The plant specific submittal will identify and address the risk acceptability of any equipment that exceeds its operating limits, conditions, and/or ratings. Any significant effects of the changes in minimum component and system performance capability or reliability will be included in the revised plant specific PRA and a description of each of these effects, as well as their quantified impacts on CDF and LERF, will be provided in the plant specific submittal.

10.5.3 Operator Response

CPPU Effect: The increase in power level results in changes to event dynamics.

CPPU Basis: CPPU reduces certain operator response times, which could decrease operator reliability. [ ] The CPPU effect will be determined when the plant specific PRA is revised and a description of each of these effects, as well as their quantified impacts on CDF and LERF, will be provided in the plant specific submittal.

10.5.4 Success Criteria

CPPU Effect: The increase in power level could have an impact on the plant PRA success criteria, which could impact the CDF and LERF results.

CPPU Basis: [ ] Any potential impact will be assessed when the plant specific PRA is revised, and a description of each of these effects, as well as their quantified impacts on CDF and LERF, will be provided in the plant specific submittal.

10.5.5 External Events

CPPU Effect: The increase in power level could have an impact on the plant PRA external events, which could impact the CDF and LERF results.

CPPU Basis: It is expected that performing an internal event PRA and addressing any resulting issues will be adequate for addressing issues related to external events PRA also. However, to address those issues that may be specific to external events, the plant specific submittal should address any vulnerabilities, outliers and anomalies that are identified in the plant's IPE external events submittal and identify how these conditions have been resolved for the CPPU power levels or demonstrate the acceptability of their risk impacts for the CPPU power levels. If the vulnerability relates to any operator actions, it should be reviewed for the CPPU conditions.

10.5.6 Shutdown Risks

CPPU Effect: The increase in power level could have an impact on the plant PRA shutdown risks, which could impact the CDF and LERF results.

CPPU Basis: The shutdown risks for BWR plants are generally low and the impact of CPPU on the CDF and LERF during shutdown is expected to be negligible. This is because there is a large inventory of water in the vessel, which provides sufficient time for taking mitigating actions. Plants with an existing shutdown PRA should revise it to reflect CPPU conditions and report the increase in CDF and LERF values in the plant specific submittal. Plants that do not have a shutdown PRA should address in the plant specific submittal the plant's shutdown risk management philosophy and controls, impacts of CPPU on shutdown conditions, and any critical, time-limited, conditions (e.g., describe the risk management process and tools used for taking systems out for maintenance during shutdown and either explain their adequacy for CPPU conditions, or address any changes made to these processes to account for changes in success criteria and time available for time-critical operations).

10.5.7 PRA Quality

CPPU Effect: CPPU has no direct effect on PRA quality.

CPPU Basis: The plant specific PRA should be of adequate quality to evaluate the impact of CPPU discussed here in Section 10.5. The plant specific submittal should address the adequacy of the plant's PRA models to reflect the as designed, as-operated plant. The plant specific submittal should also state how any weaknesses in the PRA quality identified in the staff SERs on the IPE and IPEEE submittals and any independent/peer/certification reviews, will be addressed for CPPU.

10.6 OPERATOR TRAINING AND HUMAN FACTORS

Some additional training is required to enable plant operation at the increased power level. The topics considered in this section are:

Topic	CPPU Effect	Disposition
Operator training and human factors	Power level increase	[ ]

CPPU Effect: The increase in power level results in new Technical Specifications and changes to plant performance, and new curves and actions levels in plant procedures.

CPPU Basis: [

] The operator training program is evaluated to determine the specific changes required for operator training. This evaluation includes the plant simulator.

[

] Significant events result in automatic plant shutdown (scram). Some events result in automatic reactor coolant pressure boundary pressure relief, ADS actuation and/or automatic ECCS actuation (for low water level events). All events included in the plant design basis result in safety related systems, structures and components remaining within their acceptance limits. CPPU does not change any of the automatic safety functions. After the applicable automatic responses have initiated, the follow on operator actions for plant safety (e.g., maintaining safe shutdown, core cooling, containment cooling) do not change for CPPU, although required operator response time may change.

Training required to operate the plant following uprate will be conducted prior to operation of the unit at CPPU conditions. Data obtained during uprated operation will be incorporated into additional training as needed. The classroom training will cover various aspects of CPPU, including changes to parameters, setpoints, scales, plant procedures, systems and startup test procedures. The classroom training will be combined with simulator training. The simulator training, as a minimum, will include a demonstration of transients that show the greatest change in plant response at uprate power compared to current power.

Simulator changes and fidelity revalidation will be performed in accordance with the ANSI/ANS 3.5 standard applicable to the current program.

Section 10.9 addresses the CPPU related effects on the Emergency and Abnormal Operating Procedures (EOPs).

10.7 PLANT LIFE

The plant life evaluation identifies degradation mechanisms influenced by increases in fluence and flow. The topics considered in this section are:

Topic	CPPU Effect	Disposition
Irradiated Assisted Stress Corrosion Cracking	Increased Peak Fluence	[
Flow Accelerated Corrosion	Increased Flow	]

CPPU Effect: Two degradation mechanisms are influenced by CPPU: (1) Irradiation Assisted Stress Corrosion Cracking (IASCC) and (2) Flow Accelerated Corrosion (FAC). The increase in irradiation of the core internal components influences IASCC. The increase in steam and FW flow rate influence FAC.

CPPU Basis: The longevity of most equipment is not affected by CPPU. [

] The reactor internals inspection and FAC programs will 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.

A summary of the plant specific IASCC and FAC assessments for CPPU will be reported in the plant specific power uprate submittal.

10.8 NRC AND INDUSTRY COMMUNICATIONS

For previous power uprate submittals, NRC and industry communications were reviewed to determine if a plant's pre-uprate evaluation and disposition of the communication could change due to power uprate. The topics considered in this section are:

Topic	CPPU Effect	Disposition
Plant disposition of NRC and Industry communications	Disposition not required	[ ]

CPPU Effect: Disposition not required.

CPPU Basis: 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 already are addressed in this report. In addition, all of the plant safety analyses affected by CPPU already are addressed in this report. As a result, evaluation of plant design and safety analyses affected by the communications in place inherently will be included in the plant specific CPPU assessments. Furthermore, it is GE's experience that any new safety significant issues that arise are considered in the affected system evaluations and safety analyses without the benefit of this NRC and industry communications review. 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. The topics considered in this section are:

Topic	CPPU Effect	Disposition
Emergency Operating Procedures	Values For Variables And Limits	[
Abnormal Operating Procedures	Operator Actions	]

CPPU Effect: Some of the Emergency Operating Procedures (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.

CPPU Basis: EOPs include variables and limit curves, which define conditions where operator actions are indicated. Some of these variables and limit curves depend upon the value of rated reactor power. The operator actions in the EOPs are not changed as a result of increasing rated reactor power; only the conditions at which some of the actions are specified will change. Changing some of the variables and limit curves will require modifying the values in the EOPs and updating utility support documentation. EOP curves and limits may also be included in the safety parameter display system and will be updated accordingly. The plant EOPs will be reviewed for any effects of power uprate, and the EOPs will be updated, as necessary.

AOPs include event based operator actions. Some of these operator actions may be influenced by plant modifications required to support the increase in rated reactor power. Changing some of the operator actions may require modifications to the AOPs and updating utility support documentation. The plant AOPs will be reviewed for any effects of power uprate and will be updated as necessary."

11.0 LICENSING EVALUATIONS

This section addresses the evaluations in Chapter 11 of the current plant power uprate submittals except for plant unique items, which are dispositioned in Section 10. The licensing evaluations addressed in this section include:

- Effect on Technical Specifications
- Environmental Assessment
- Significant Hazards Consideration Assessment

11.1 EFFECT ON TECHNICAL SPECIFICATIONS

Implementation of CPPU requires revision of a number of the Technical Specifications. A generic list of Technical Specifications that could be affected by a CPPU has been developed and is provided in Table 11-1. Also included in this list are Technical Specifications that are referenced to Rated Thermal Power (RTP); however, some of these do not require change. Each Technical Specifications item in this list is based upon the content of the improved Standard Technical Specifications (References 11 and 12) and identifies: (1) the potential for requiring any change, (2) a description of each item, and (3) the disposition of the change, including a cross reference to sections in the report or Appendix A that support the change. This list will be used as guidance for the development of the plant unique Technical Specifications changes to be requested by a utility. However, additional Technical Specifications changes may be identified based on a review of the plant specific Technical Specifications and related changes requested on a plant unique basis.

11.2 ENVIRONMENTAL ASSESSMENT

Each license amendment request will have its own environmental assessment. The following is generic input to this assessment for CPPU. Plant specific assessments may reference all or a part of the following. These plant specific assessments will accompany the plant specific submittal.

The environmental effects of CPPU will be controlled at the same limits as for the current analyses. Normally, none of the present limits for plant environmental releases will be increased as a consequence of uprate. Nonradioactive environmental discharges increase very slightly due to CPPU. Liquid discharges may be slightly warmer and/or have small increases in dissolved and suspended solids. There is essentially no change in the non-radiological atmospheric releases.

CPPU has no significant effect on the nonradiological elements of concern, and the plant will be operated in an environmentally acceptable manner as established by the Final Environmental Statement. Existing Federal, State and local regulatory permits presently in effect will usually accommodate CPPU without modification. The makeup water sources requirements are not increased beyond the present Environmental Protection Plan. Effects to air, water, and land resources are nonexistent.

The evaluation of effects of CPPU on radiological effluents or offsite doses is summarized in Section 8. There may be very slight increases in the radionuclides released to the environment through gaseous and liquid effluents, but well within design and regulatory limits. This will be confirmed in the plant specific submittal. The quantity of spent fuel will not be significantly affected by the uprate. The short-term radioactivity level will be slightly higher, but still below the previously established limits. The effect of CPPU will be insignificant, subject to the above confirmatory check, and the normal effluents and doses will remain well within 10CFR20 and 10CFR50, Appendix I limits.

For plants with a cooling tower, operation at CPPU will require slightly increased cooling tower makeup water flow due to expected changes in tower evaporation and potential system blowdown. Accordingly, intake velocities at the intake structure to the plant will change slightly.

The proposed CPPU does not require a change to the Environmental Protection Plan or constitute an unreviewed environmental question because it does not involve:

- A significant increase in any adverse environmental effect previously evaluated in the final statement, environmental effect appraisals, or in any decisions of the Atomic Safety and Licensing Board; or
- A significant change in effluents; or
- A matter not previously reviewed and evaluated in the documents specified above which may have a significant adverse environmental effect.

The evaluations also establish that CPPU qualifies for a categorical exclusion not requiring an environmental review in accordance with 10CFR51.22(c)(9) because it does not:

- Involve a significant hazard, or
- Result in a significant increase in the amounts of any effluents that may be released offsite; or
- Result in a significant increase in individual or cumulative occupational radiation exposure.

11.3 SIGNIFICANT HAZARDS CONSIDERATION ASSESSMENT

Each license amendment request will have its own significant hazards consideration assessment. The following is generic input to this significant hazards assessment for CPPU. Plant specific assessments may reference all or a part of the following. These plant specific assessments will accompany the plant specific submittal.

Increasing the power level of nuclear power plants while maintaining the reactor pressure can be done safely within plant specific limits, and is a highly cost effective way to increase the installed electricity generating capacity.

The power uprate submittal will provide all significant safety analyses and evaluations to justify increasing the licensed thermal power up to 120% of the Original Licensed Thermal Power (OLTP).

11.3.1 Modification Summary

An increase in electrical output of a BWR plant is primarily accomplished by generation and supply of higher steam flow to the turbine generator. Continuing improvements in the analytical techniques (computer codes and data) based on several decades of BWR safety technology, plant performance feedback, and improved fuel and core designs have resulted in a significant increase in the design and operating margins between calculated safety analysis results and the licensing limits. These available safety analysis improvements, combined with the excess as-designed equipment, system and component capabilities, provide BWR plants the capability to increase their thermal power ratings with no significant increase in the hazards presented by the plant as approved by the NRC at the original license stage. An increase in the thermal power rating of up to 20% can be usually accomplished without major Nuclear Steam Supply System (NSSS) hardware modifications, and can be done with limited non-safety hardware modifications.

The plan for achieving higher power is to expand the power flow map by extending the standard Maximum Extended Load Line Limit Analysis (MELLLA) upper boundary and the maximum core flow line to the uprated power. However, there is no increase in the maximum core flow or operating pressure over the pre-uprate values. For CPPU operation, the plant already has or can readily be modified to have adequate control over inlet pressure conditions at the turbine, to account for the larger pressure drop through the steam lines at higher steam flow and to provide sufficient pressure control and turbine flow capability.

11.3.2 Discussions of Issues Being Evaluated

Plant performance and responses to hypothetical accidents and transients have been analyzed for a power uprate license amendment. This section summarizes the safety significant plant reactions to events analyzed for licensing the plant, and the potential effects on various margins of safety, and thereby concludes that no significant hazards consideration will be involved.

11.3.2.1 Uprate Analysis Basis

The CPPU safety analyses are based on a Regulatory Guide 1.49 power factor times the uprated power level, except for some analyses that are performed at nominal uprated power, either because the Regulatory Guide 1.49 power factor is already accounted for in the analysis methods or Regulatory Guide 1.49 does not apply (e.g., ATWS and SBO events).

11.3.2.2 Margins

The above CPPU safety analysis basis ensures that the power dependent margins prescribed by the Code of Federal Regulations (CFR) are maintained by meeting the appropriate regulatory criteria. NRC-accepted computer codes and calculational techniques are used for the evaluations that demonstrate meeting the acceptance criteria. Similarly, design margins specified by application of the American Society of Mechanical Engineers (ASME) design rules are maintained, as are other margin ensuring criteria used to judge the acceptability of the plant. Environmental margins are maintained by not increasing any of the present limits for releases.

11.3.2.3 Fuel Thermal Limits

No change is required in the mechanical fuel design to achieve the CPPU or to meet the plant licensing limits. No increase in allowable peak bundle power is requested for CPPU. The current fuel design limits will still be met at the uprated power level. Analyses for each fuel reload will continue to meet the criteria accepted by the NRC as specified in Reference 3 or otherwise approved in the Technical Specification amendment request. In addition, future fuel designs will meet acceptance criteria approved by the NRC.

11.3.2.4 Makeup Water Sources

The BWR design concept includes a variety of ways to pump water into the reactor vessel to deal with all types of events. There are numerous safety related and non-safety related cooling water sources. The safety related cooling water sources alone maintain core integrity by providing adequate cooling water. There are high and low pressure, high and low volume, safety and non-safety grade means of delivering water to the vessel. These means include at least:

- Feedwater and condensate system pumps
- Low pressure emergency core cooling system (LPCI & CS/LPCS) pumps
- High pressure emergency core cooling system (HPCI or HPCS) pump
- Reactor core isolation cooling (RCIC) pump
- Standby liquid control (SLC) pumps
- Control rod drive (CRD) pumps.

Many of these diverse water supply means are redundant in both equipment and systems.

CPPU does not result in an increase or decrease in the available water sources, nor does it change the selection of those assumed to function in the safety analyses. NRC-approved methods were used to evaluate the performance of the Emergency Core Cooling Systems (ECCS) during postulated Loss-Of-Coolant Accidents (LOCA).

CPPU results in an increase in decay heat and, thus, the core cooling time to reach cold shutdown requires more time. However, this is not a safety concern, and the existing cooling capacity can bring the plant to cold shutdown within an acceptable time span.

11.3.2.5 Design Basis Accidents

Design Basis Accidents (DBAs) are very low probability hypothetical events whose characteristics and consequences are used in the design of the plant, so that the plant can mitigate their consequences to within acceptable regulatory limits. For BWR licensing evaluations, capability is demonstrated for coping with the range of hypothetical pipe break sizes in the largest recirculation, steam, and feedwater lines, a postulated break in one of the ECCS lines, and the most limiting small lines. This break range bounds the full spectrum of large and small, high and low energy line breaks; and demonstrates the ability of plant systems to mitigate the accidents while accommodating a single active equipment failure in addition to the postulated LOCA. Several of the most significant licensing assessments are based on the LOCA and include:

- Challenges to Fuel (ECCS Performance Analyses) (Regulatory Guide 1.70 and SAR Section 6.3) in accordance with the rules and criteria of 10CFR50.46 and Appendix K where the limiting criterion is the fuel Peak Clad Temperature (PCT).
- Challenges to the Containment (Regulatory Guide 1.70 and SAR Section 6.2) wherein the primary criteria of merit are the maximum containment pressure calculated during the course of the LOCA and maximum suppression (cooling) pool temperature for long-term cooling in accordance with 10CFR50 Appendix A Criterion 38.
- DBA Radiological Consequences (Regulatory Guide 1.70 and SAR Section 15) calculated and compared to the criteria of 10 CFR 100, 10 CFR 50.67, 10 CFR 50, Appendix A GDC-19, or plant specific limits.

11.3.2.6 Challenges to Fuel

Emergency Core Cooling Systems are described in Section 6.3 of the plant Updated Final Safety Analysis Report (UFSAR). CPPU will have only a minor effect on the PCT consequences of a LOCA. The ECCS performance evaluation demonstrates the continued conformance to the acceptance criteria of 10CFR50.46. The licensing safety margin is not affected by CPPU. The increased PCT consequences for CPPU are insignificant compared to the amount by which the results are below the regulatory criteria. Therefore, the ECCS safety margin is not significantly affected by CPPU.

11.3.2.7 Challenges to the Containment

The CPPU peak values for containment pressure and temperature meet regulatory requirements and, therefore, confirm the suitability of the plant for operation at uprated power. The effect of CPPU on the conditions that affect the containment dynamic loads also meet requirements. Where plant conditions with CPPU are within the range of conditions used to define the current dynamic loads, current safety criteria are met and no further structural analysis is required. Otherwise, the structure was evaluated to ensure that the safety criteria are met. The change in short-term containment response is negligible. Because there is more residual heat with CPPU, the containment long-term response is slightly more severe. However, containment pressures and temperatures remain below their design limits following any DBA, and, thus, the containment and its cooling systems are judged to be satisfactory for CPPU operation.

11.3.2.8 Design Basis Accident Radiological Consequences

The magnitude of the potential radiological consequences is dependent upon the quantity of fission products released to the environment, the atmospheric dispersion factors and the dose exposure pathways. The atmospheric dispersion factors and the dose exposure pathways do not change. Therefore, the only factor that could influence the magnitude of the consequences is the quantity of activity released to the environment. This quantity is a product of the activity released from the core and the transport mechanisms between the core and the effluent release point.

The radiological consequences of LOCA inside containment, Main Steam Line Break Accident (MSLBA) outside containment, Instrument Line Break Accident (ILBA), Control Rod Drop Accident (CRDA) and Fuel Handling Accident (FHA) are reevaluated for CPPU. The radiological results for all accidents remain below the applicable limits for the plant.

11.3.2.9 Anticipated Operational Occurrence Analyses

Anticipated Operational Occurrences (AOOs) are evaluated against the Safety Limit Minimum Critical Power Ratio (SLMCPR). The SLMCPR is determined using NRC-approved methods. The most limiting transient is slightly more severe when initiated from the uprate power level, and may result in a slightly larger change in CPR than that initiated from the current power level. The limiting transients are core specific and are analyzed for each reload fuel cycle. Licensing acceptance criteria will be met. Therefore, the margin of safety is not affected by CPPU.

11.3.2.10 Combined Effects

CPPU analyses use fuel designed to current NRC-approved criteria and the plant is operated within NRC-approved limits to produce more power in the reactor, and thus, increases steam flow to the turbine. NRC-approved design criteria are used to ensure equipment mechanical performance safety at uprated conditions. Scram frequency is maintained by small adjustments to reactor instrumentation. These adjustments are attributed to the small changes in the reactor operating conditions. DBAs are hypothesized to evaluate challenges to the fuel, containment and off-site dose limits. These challenges are evaluated separately in accordance with conservative regulatory procedures such that the separate effects are more severe than any combined effects. The off-site dose evaluation specified by Regulatory Guide 1.3 and SRP-15.6.5 provides a more severe DBA radiological consequences scenario than the combined effects of the hypothetical LOCA, which produces the greatest challenge to the fuel and/or containment. That is, the DBA, which produces the highest PCT and/or containment pressure, does not damage large amounts of fuel, and thus, the source terms and doses are much smaller than those postulated in conformance with Regulatory Guide 1.3 evaluations.

11.3.2.11 Non-LOCA Radiological Release Accidents

All of the other radiological releases discussed in Regulatory Guide 1.70 and UFSAR Chapters 11 and 15 are either unchanged because they are not power-dependent, or increase at most by the amount of the uprate.

11.3.2.12 Equipment Qualification

Plant equipment and instrumentation have been evaluated against the applicable criteria. Significant groups/types of the equipment have been justified for CPPU by generic evaluations. Some of the qualification testing/justification at the current power level was done at more severe conditions than the minimum required. In some cases, the qualification envelope did not change significantly due to power uprate. Where the qualification envelope changes, the equipment or instrumentation will be evaluated to assure their acceptability for the new environment.

11.3.2.13 Balance-of-Plant

Balance-Of-Plant (BOP) systems/equipment used to perform safety related and normal operation functions have been reviewed for CPPU in a manner comparable to that for safety related NSSS systems/equipment. This included, but was not necessarily limited to, all or portions of the main steam, feedwater, turbine, condenser, condensate, essential and non-essential service water, emergency diesel generator, BOP piping, and support systems.

11.3.2.14 Environmental Consequences

The environmental effects of CPPU will be controlled below the same limits as for the current power level. That is, none of the present environmental release limits are increased as a result of CPPU. A management procedure will be in place for all environmental limits with which the plant is presently required to comply. The current environmental release margins are thereby maintained.

11.3.2.15 Technical Specifications Changes

The Technical Specifications ensure that plant and system performance parameters are maintained within the values assumed in the safety analyses. That is, the Technical Specifications parameters (setpoints, allowable values, operating limits, etc.) are selected such that the actual equipment is maintained equal to or more conservative than the assumptions used in the safety analyses. The improved Standard Technical Specifications that could be affected by CPPU are listed in Table 11-1. Plant specific Technical Specifications changes are provided with the plant specific submittal. Proper account is taken for inaccuracies introduced by instrument drift, instrument accuracy, and calibration accuracy. This ensures that the actual plant responses at uprated condition are less severe than those represented by the safety analysis. Similarly, the Technical Specifications address equipment operability (availability) and put limits on equipment out-of-service (not available for use) times such that the plant can be expected to have at least the complement of equipment available to mitigate abnormal plant events assumed in the safety analyses. Because the safety analyses for CPPU show that the results are acceptable within regulatory limits, there is no undue risk to public health and safety. Technical Specifications changes consistent with the CPPU level are made in accordance with methodology approved for the plant and continue to provide a comparable level of protection as Technical Specifications previously issued by the NRC.

11.3.3 Assessment of 10CFR50.92 Criteria

10CFR50.91(a) states "At the time a licensee requests an amendment, it must provide to the Commission its analysis about the issue of no significant hazards consideration using the standards in §50.92." The following provides this analysis for CPPU up to 120% of the original licensed thermal power.

1) Will the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The increase in power level discussed herein will not significantly increase the probability or consequences of an accident previously evaluated.

The probability (frequency of occurrence) of DBA occurring is not affected by the increased power level, because the plant still complies with the regulatory and design basis criteria established for plant equipment (ASME code, IEEE standards, NEMA standards, Reg. Guide criteria, etc.). An evaluation of the BWR probabilistic safety assessments concludes that the calculated core damage frequencies do not significantly change due to Constant Pressure Power Uprate (CPPU). Scram setpoints (equipment settings that initiate automatic plant shutdowns) are established such that there is no significant increase in scram frequency due to power uprate. No new challenge to safety related equipment results from CPPU.

The changes in consequences of hypothetical accidents, which would occur from 102% of uprated power compared to those previously evaluated, are in all cases insignificant. The CPPU accident evaluations do not exceed any of their NRC-approved acceptance limits. The spectrum of hypothetical accidents and abnormal operational occurrences has been investigated, and are shown to meet the plant's currently licensed regulatory criteria. In the area of core design, for example, the fuel operating limits such as Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) and Safety Limit Minimum Critical Power Ratio (SLMCPR) are still met, and fuel reload analyses will show plant transients meet the criteria accepted by the NRC as specified in Reference 3. Challenges to fuel (ECCS performance) are evaluated, and shown to still meet the criteria of 10CFR50.46 and Appendix K, and Regulatory Guide 1.70 SAR Section 6.3. Challenges to the containment have been evaluated, and the containment and its associated cooling systems meet 10CFR50 Appendix A Criterion 38, Long Term Cooling, and Criterion 50, Containment. Radiological release events (accidents) have been evaluated, and meet the criteria of 10CFR100, 10CFR50.67, 10CFR50, Appendix A GDC-19, or plant specific limits.

2) Will the change create the possibility of a new or different kind of accident from any accident previously evaluated?

As summarized below, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Equipment that could be affected by CPPU has been evaluated. No new operating mode, safety related equipment lineup, accident scenario or equipment failure mode was identified. The full spectrum of accident considerations, defined in Regulatory Guide 1.70, has been evaluated, and no new or different kind of accident has been identified. CPPU uses already developed technology, and applies it within the capabilities of already existing plant equipment in accordance with presently existing regulatory criteria to include NRC approved codes, standards and methods.

3) Will the change involve a significant reduction in a margin of safety?

As summarized below, this change will not involve a significant reduction in a margin of safety.

The calculated loads on all affected structures, systems and components have been shown to remain within their design allowables for all design basis event categories. No NRC acceptance criterion is exceeded. Only some design and operational margins are affected by CPPU. The margins of safety currently designed into the plant are not affected by CPPU. Because the plant configuration and reactions to transients and hypothetical accidents do not result in exceeding the presently approved NRC acceptance limits, CPPU does not involve a significant reduction in a margin of safety.

Conclusions:

A CPPU up to 120% of original licensed thermal power has been investigated. The method for achieving higher power is to slightly increase some plant operating parameters. The plant licensing challenges have been evaluated and it has been demonstrated that 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 presently existing regulatory limits or acceptance criteria applicable to the plant, which might cause a reduction in a margin of safety.

Having arrived at negative declarations with regards to the criteria of 10CFR50.92, this assessment concludes that a CPPU up to 120% of the original licensed thermal power described herein does not involve a Significant Hazards Consideration.

Table 11-1 Potential Technical Specifications Changes

Potentially Affected ITS Section (Ref. 9 & 10)	Potential TS Change	Description	Disposition
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Potentially Affected ITS Section (Ref. 9 & 10)	Potential TS Change	Description	Disposition
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Potentially Affected ITS Section (Ref. 9 & 10)	Potential TS Change	Description	Disposition
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Potentially Affected ITS Section (Ref. 9 & 10)	Potential TS Change	Description	Disposition
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Potentially Affected ITS Section (Ref. 9 & 10)	**Potential TS Change**	**Description**	**Disposition**
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Potentially Affected ITS Section (Ref. 9 & 10)	Potential TS Change	Description	Disposition
[			
			]

Potentially Affected ITS Section (Ref. 9 & 10)	Potential TS Change	Description	Disposition
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12.0 REFERENCES

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