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GE Nuclear Energy

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LICENSING TOPICAL REPORT

CONSTANT PRESSURE POWER UPRATE

Approved: Christina C. Roberts

Christina C. Roberts, Manager
BWR Asset Enhancement Services

U.S. NUCLEAR REGULATORY COMMISSION
In the Matter of Entergy Nuclear Vermont Yankee L.L.C.
Docket No. 50-271 Official Exhibit No. Entergy 25
OFFERED by: Applicant/Licensee Intervenor (Staff 12,13)
NRC Staff Other
IDENTIFIED on 9/13/06 Witness/Panel Nichols/Casillas
Action Taken: ADMITTED REJECTED WITHDRAWN
Reporter/Clerk MAC

Template=SECY-028

SECY-02

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This is a non-proprietary version of the document NEDC-33004P-A, Revision 4,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

PLEASE READ CAREFULLY

The information contained in this document is furnished for the purpose of obtaining NRC approval of the licensing requirements to increase Boiling Water Reactor licensed thermal power up to 120% of original thermal power while holding the reactor dome pressure constant. The only undertakings of General Electric Company respecting information in this document are contained in the contracts between General Electric Company and the participating utilities in effect at the time this report is issued, and nothing contained in this document shall be construed as changing those contracts. The use of this information by anyone other than that for which it is intended is not authorized; and with respect to any unauthorized use, General Electric Company makes no representation or warranty, and assumes no liability as to the completeness, accuracy, or usefulness of the information contained in this document.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 31, 2003

MFN:03-027

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GENERAL ELECTRIC COMPANY

Mr. James F. Klapproth, Manager
Engineering & Technology
GE Nuclear Energy
175 Curtner Ave
San Jose, CA 95125

SUBJECT: REVIEW OF GE NUCLEAR ENERGY LICENSING TOPICAL REPORT, NEDC-33004P, REVISION 3, "CONSTANT PRESSURE POWER UPRATE" (TAC NO. MB2510)

Dear Mr. Klapproth:

By letter dated February 6, 2003, which incorporates Errata and Addenda 1, GE Nuclear Energy (GENE) provided a revised Constant Pressure Power Uprate (CPPU) licensing topical report (LTR) documenting the basis for the approach to be used for GE-prepared BWR constant pressure power uprate safety analysis reports. The revision addresses NRC's concerns stated in a letter dated August 12, 2002, withdrawing our safety evaluation (SE) approving the use of the CLTR. The NRC has considered this request and with the exception of the proposed elimination of large transient testing, has approved the use of this LTR. The staff intends to issue a supplement to the SE when the large transient testing guidance is available.

Licensees proposing to reference the CPPU LTR 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 mentioned in the seven restrictions applicable to the CPPU LTR must first request and obtain a license amendment for the associated change in accordance with the CPPU LTR. The one exception is with regards to a source term methodology change. A licensee may submit and the NRC staff will review a source term methodology change, in lieu of the analysis in Section 9.2 of the CPPU LTR, concurrent with the power uprate request, if the source term submittal supports operation at the uprated power level. Licensees proposing to utilize fuel designs other than GE fuel, up through GE 14 fuel, may not reference the CPPU LTR as a basis for their power uprate since the CPPU LTR process applies only to GE fuel and GE accident analysis methods. However, such licensees may reference the CPPU LTR for areas other than those involving reactor systems and fuel issues which are not impacted by the fuel design. Licensees should afford the staff sufficient time to complete its review of all associated licensing basis changes prior to submittal or request for the implementation of the power uprate when referencing the CPPU LTR.

The staff finds that the subject topical report is acceptable for referencing in licensing applications to the extent specified under the limitations delineated in the report and in the associated NRC safety evaluation. The enclosed safety evaluation defines the basis for acceptance of the CPPU LTR. As discussed in Section 10.5 of the safety evaluation, the staff is preparing guidance to generically address the requirement for conducting large transient tests in conjunction with power uprates. Therefore, the staff is not prepared to accept, on a generic basis, the proposed elimination of these tests. The staff intends to issue a supplement to this safety evaluation when the guidance is available.

We do not intend to repeat our review of the matters described in the subject report, and found acceptable, when the report appears as a reference in license applications, except to ensure that the material presented applies to the specific plant involved. Our acceptance applies only to matters approved in the report.

In accordance with the guidance provided on the NRC website, we request that GENE publish an accepted version within three months of receipt of this letter. The accepted version shall incorporate (1) this letter and the enclosed SE between the title page and the abstract, and (2) a "-A" (designating "accepted") following the report identification symbol.

If the NRC's criteria or regulations change so that its conclusion in this letter that the topical report is acceptable is invalidated, GENE and/or the applicant referencing the topical report will be expected to revise and resubmit its respective documentation, or submit justification for the continued applicability of the topical report without revision of the respective documentation.

Pursuant to 10 CFR 2.790, we have determined that the safety evaluation provided as Enclosure 1 contains proprietary information. Proprietary information contained in Enclosure 1 is indicated by double underlines. We have prepared a non-proprietary version of the safety evaluation (Enclosure 2) that we have determined does not contain proprietary information. However, we will delay placing Enclosure 2 in the public document room for a period of ten (10) working days from the date of this letter to provide you with the opportunity to comment on the proprietary aspects only. If you believe that any information in Enclosure 2 is proprietary, please identify such information line by line and define the basis pursuant to the criteria of 10 CFR 2.790.

If you have any questions, please contact Alan Wang, GENE Project Manager, at (301) 415-1445.

Sincerely,

William H. Ruland

William H. Ruland, Director
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 710

Enclosures: 1. Proprietary Safety Evaluation
2. Non-Proprietary Safety Evaluation

cc w/encl. 2: See next page

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TABLE OF CONTENTS

LIST OF ACRONYMS

AC	- alternating current
ACP	- activated corrosion products
ADS	- automatic depressurization system
ALARA	- as low as is reasonably achievable
ANSI	- American National Standards Institute
AOO	- anticipated operational occurrence
AOP	- abnormal operating procedure
APRM	- average power range monitor
ARI	- alternate rod insertion
ART	- adjusted reference temperature
ASME	- American Society of Mechanical Engineers
AST	- alternate source term
ATWS	- anticipated transient without scram
AV	- allowable value
BOP	- balance of plant
BHP	- brake horse power
BWR	- boiling water reactor
BWRVIP	- Boiling Water Reactor Vessel and Internals Project
CCW	- component cooling water
CDF	- core damage frequency
CFDS	- condensate filtration and demineralization system
CGCS	- combustible gas control system
COLR	- Core Operating Limits Report
CPPU	- constant pressure power uprate
CRD	- control rod drive
CRDA	- control rod drop accident
CRDM	- control rod drive mechanism
CS	- containment spray
CSC	- containment spray cooling
CST	- condensate storage tank
DBA	- design-basis accident
DC	- direct current
ECCS	- emergency core cooling system
EMA	- equivalent margins analysis
EOP	- emergency operating procedure
EPU	- extended power uprate
EQ	- environmental qualification
ESFAS	- engineered safety feature actuation system
FAC	- flow accelerated corrosion
FHA	- fuel handling accident
FIV	- flow induced vibration
FW	- feedwater
GDC	- general design criteria
GENE	- GE Nuclear Energy
GESTAR II	- General Electric Standard Application for Reactor Fuel
GL	- generic letter

GNF	- Global Nuclear Fuel
HCU	- hydraulic control unit
HELB	- high energy line break
HPCI	- high pressure coolant injection
HPCS	- high pressure core spray
HWC	- hydrogen water chemistry
IASCC	- irradiation assisted stress corrosion cracking
ICA	- interim corrective action
ILBA	- instrument line break accident
IORV	- inadvertent opening of a relief valve
IPE	- individual plant examination
IPEEE	- individual plant examination of external events
LTR	- licensing topical report
LHGR	- linear heat generation rate
LOCA	- loss of coolant accident
LTS	- long term solution
LOFW	- loss of feedwater
LOFWF	- loss of feedwater flow
LPCI	- low pressure coolant injection
LPCS	- low pressure core spray
LERF	- large early release frequency
LOOP	- loss of offsite power
MELLLA	- maximum extended load limit line analysis
MEOD	- maximum extended operating domain
MCPR	- minimum critical power ratio
MAPLHGR	- maximum average planar linear heat generation rate
MSLBA	- main steam line break accident
MSIV	- main steam isolation valve
MVAR	- reactive power
MOS	- main offgas system
NSSS	- nuclear steam supply system
NPSH	- net positive suction head
NMIP	- noble metals injection process
OLTP	- original licensed thermal power
OLMCPR	- operating limit minimum critical power ratio
OPRM	- oscillation power range monitor
PUSAR	- power uprate safety analysis report
PRFO	- pressure regulatory failure to open
P-T	- pressure temperature
PCT	- peak cladding temperature
PRA	- probabilistic risk assessment
RPT	- reactor pressure temperature
RHR	- residual heat removal
RCS	- reactor coolant system
RIPD	- reactor internal pressure differences
RPV	- reactor pressure vessel
RCPB	- reactor coolant pressure boundary
RCIC	- reactor core isolation cooling
RWCU	- reactor water cleanup

RCIS	- rod control and information system
RTP	- rated thermal power
RWL	- rod withdrawal limiter
RPS	- reactor protection system
SE	- safety evaluation
SRLR	- Supplemental Reload Licensing Report
SAFDL	- specified acceptable fuel design limit
SLMCPR	- safety limit minimum critical power ratio
SBO	- station blackout
SRV	- safety relief valve
SRP	- Standard Review Plan
SCCR	- spent condensate cleanup resins
SLC	- standby liquid control
SPC	- suppression pool cooling
SDC	- shutdown cooling
SGTS	- standby gas treatment system
STS	- standard technical specifications
TAF	- top-of-active fuel
TTNBP	- turbine trip no bypass
TCV	- turbine control valve
TS	- technical specification
UFSAR	- updated final safety analysis report
USE	- upper shelf energy

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

GE NUCLEAR ENERGY LICENSING TOPICAL REPORT

NEDC-33004P, REVISION 1

"CONSTANT PRESSURE POWER UPRATE"

PROJECT NO. 710

1.0 OVERVIEW

1.1 Introduction

By letter dated March 19, 2001 (Reference 1), GE Nuclear Energy (GENE) submitted Licensing Topical Report (LTR) NEDC-33004P, "Constant Pressure Power Uprate" (CPPU) for NRC review and approval. The CPPU LTR proposes a simplified process for achieving extended power uprates (EPU). Following meetings with the NRC staff on June 13, 2001, and July 17, 2001, GENE submitted Revision 1 of the CPPU LTR by letter dated July 26, 2001 (Reference 2). After initial staff review, a meeting was held on September 26, 2001, to discuss staff questions. The staff review of the CPPU LTR was performed in conjunction with ongoing reviews and audits of recent EPU requests that used elements of the constant pressure uprate approach. GENE submitted responses to the staff's request for additional information (RAI) on December 3, 18, and 21, 2001 (References 3, 4, and 5). GENE also submitted an update (errata and Addenda 1) to the CPPU LTR on December 21, 2001 (Reference 6) to provide consistency with the RAI responses and to provide additional improvements and corrections which summarized discussions with the NRC staff. The CPPU LTR submittal and the supplemental supporting documentation regarding the revised constant pressure EPU approach have been reviewed by the staff. A portion of the staff's review was performed during on-site audits of the GENE/Global Nuclear Fuel (GNF) safety analyses performed for power uprate safety analysis reports (PUSAR) supporting recent licensee EPU amendment requests.

On June 20, 2002, the staff issued its safety evaluation (SE) regarding NEDC-33004P, Revision 1 to GENE with a 10-day hold before release to the public. We informed you that this was to provide GENE an opportunity to comment on the proprietary aspects before making the SE public. In subsequent calls with your staff, GENE informed the NRC staff that they had substantive concerns regarding the content of the SE. By letter dated July 18, 2002, GENE formally documented their comments regarding the proprietary and technical content of the SE.

During a meeting with the Browns Ferry licensee on July 10, 2002, the staff learned that GENE and TVA intended to apply the CPPU LTR in a way that did not reflect the NRC staff's understanding and basis for the acceptability of the topical report for licensing applications. In telephone discussions with GENE on July 15 and 22, 2002, the staff confirmed that our understanding about the manner in which the CPPU topical report could be applied and the restrictions on the use of the CPPU topical report is significantly different from GENE's.

The nature of the misunderstanding involved the degree to which a licensee can pursue changes to the licensing basis in parallel with a power uprate. To achieve this goal, the report describes, on page 1-1, a number of changes to a plant's licensing basis which are excluded from consideration as part of the CPPU process. However, this list is preceded by a sentence which GENE believes allows those changes to be reviewed and implemented, in parallel with the CPPU process, as separate licensing actions.

The NRC staff did not agree with this interpretation. The exclusions listed on page 1-1 of NEDC-33004P, Revision 1, are integral to the staff's technical basis for approving the topical report. These exclusions provided the basis to allow the CPPU review to focus on issues that relate only to the power uprate itself. On August 12, 2002, we withdrew our SE that found the proposed approach in NEDC-33004P, Revision 1, to be acceptable for referencing in licensing applications. In our August 12, 2002, letter we noted that we believed that your position would have significantly reduce the efficiency gains of applying this topical report by requiring the staff to conduct extensive plant-specific analyses of the CPPU uprate along with the reload analysis. In addition, we stated that we believed that the CPPU process is viable but that GENE needed to revise and resubmit the topical report to remove the described ambiguities.

On September 10, 2002, the staff met with GENE. During this meeting GENE agreed to revise the CPPU LTR to address the use of fuel manufactured by other vendors. GENE submitted an update (errata and Addenda 2) to the CPPU LTR on October 7, 2002 (Reference 16). This submittal proposed six scenarios for the application of the CPPU LTR. On October 7, 2002, the staff had a conference call to discuss the proprietary aspects of the CPPU LTR. GENE agreed that revised CPPU LTR was needed to address the proprietary issues and several open technical issues. The staff has also had numerous technical conference calls regarding the CPPU LTR. By letter dated November 15, 2002, GENE withdrew all previous versions of the CPPU LTR and their request for proprietary protection. In addition, Revision 2 of the CPPU LTR, in a redacted form, was attached to this letter. Subsequently, the staff had a call with GENE on January 17, 2003, to discuss the November 15, 2002, revision. While it did provide some additional clarity to GENE's CPPU approach, the staff could not come to agreement with GENE on their approach. The staff stated that licensees proposing to reference this LTR as a basis for a power uprate license amendment request, and also, proposing to obtain a license amendment to incorporate one or more of the plant changes listed below must first request and obtain a license amendment for the associated change prior to the start of the staff review of the power uprate request that references this topical report. GENE requested a meeting to discuss the basis for this restriction and how the LTR would need to be revised to reflect this. On January 22, 2003, GENE met with the staff. GENE revised the LTR to reflect an approach that would provide the staff reasonable assurance that the CPPU LTR could be used as a reference for future licensing activities. By letter dated February 6, 2003, GENE submitted Revision 3 of the CPPU LTR (Reference 17).

1.2 Background

An increase in the electrical output of a boiling water reactor (BWR) is accomplished primarily by supplying a higher steam flow to the turbine generator. Most GE BWRs, as originally built and licensed, have as-designed equipment and system capability to accommodate steam flow rates at least 5 percent above the original rated power. In addition, improved analytical techniques and computer codes, plant operating experience, and improved fuel designs have resulted in a significant increase in the design and operating margins between the results of the safety analysis calculations and the licensing limits. The increased margins combined with the as-designed excess equipment, system, and component capabilities have allowed many BWRs to increase their thermal power ratings by 5 percent (stretch uprate) without modifying any nuclear steam supply system (NSSS) hardware and to increase power by up to 20 percent (extended power uprate) with some hardware modifications.

1.2.1 Extended Power Uprate (EPU) Approach

The EPU LTR (Reference 7), known as ELTR1, contains a set of generic guidelines to be met and a general approach to be followed for plants that planned extended reactor thermal power uprates of up to 120 percent of their original licensed thermal power (OLTP). These guidelines and the subsequent ELTR2 submittal (Reference 8) of generic evaluations had been developed based on the expectation that the maximum reactor operating pressure would also need to be increased to achieve the extended power uprate. These generic guidelines and generic evaluations, together with the associated NRC staff position paper and safety evaluation (SE) [incorporated in ELTR1 and ELTR2 (References 9 and 10)] have been applied to all extended power uprate submittals since their NRC review and acceptance or endorsement for referencing.

The approach to achieving an EPU consists of:

- an increase in the core thermal power with a more uniform power distribution achieved by better fuel management techniques to create increased steam flow,
- a corresponding increase in the feedwater system flow,
- no increase in maximum core flow, and
- reactor operation primarily along the maximum extended load line limit analysis (MELLLA) or maximum extended operating domain (MEOD) rod/flow lines.

The ELTR2 generic evaluations assume:

- a 20 percent increase in the thermal power,
- an increase in operating dome pressure up to 1,095 psia,
- a reactor coolant temperature increase to 556°F, and
- a steam and feedwater flow increase of about 24 percent.

For some CPPU evaluations, bounding analyses and evaluations provided in ELTR2 will be cited; however, there are restrictions on plant changes that are allowed under CPPU, most notably, that the existing maximum plant operating pressure must be maintained.

In general, the generic system and equipment performance analyses and the generic transient and accident analyses documented in ELTR1 and ELTR2 are applicable to the CPPU approach. Exceptions and deviations to generic ELTR1 and ELTR2 conclusions are identified in individual sections of the CPPU LTR and are evaluated in the corresponding sections of this SE.

1.2.2 CPPU Approach

As a result of experience in implementing EPU and licensee feedback, GENE developed a simplified approach to achieving uprated reactor power. This approach maintains the current plant maximum reactor operating pressure, and has other significant restrictions on plant changes. Extended power uprates without a reactor pressure increase have now been reviewed, approved and utilized at several plants. This CPPU approach is expected to be used by GENE as the primary basis for the majority of future extended power uprate applications. The GENE current experience base with power uprates is provided in Table 1-1 of the CPPU LTR. By achieving a power uprate without a reactor pressure increase, there can be a significant reduction in calculations and effort required compared to the EPU safety analysis and system performance evaluations. This constant pressure constraint, along with other required limitations and restrictions discussed in the CPPU LTR, allows a simplified approach to power uprate analyses and evaluations.

The method for achieving higher power at GE BWRs is to retain the MELLLA or MEOD power/flow map, and to increase core flow (and power) along the existing flow control rod line. The proposed CPPU approach will not increase the reactor operating pressure or the current licensed maximum core flow. CPPU operation will not require an increase in reactor vessel dome pressure because the plant will make modifications to the power generation equipment, pressure controls and turbine flow capabilities to control the pressure at the turbine inlet.

The CPPU LTR documents the approach to be followed to provide the basis for future CPPU applications. The overall evaluation and analyses approach has been simplified to take advantage of the constant pressure assumption. In addition, further experience with previous extended power uprate applications, more recent generic evaluations, and the use of approved reload core analysis and a standard reload licensing process are cited by GENE. These factors have been incorporated into the overall approach to simplify the required plant-specific documentation while maintaining a systematic licensing and safety evaluation process. Further, the focus of the evaluation has been placed on the safety evaluations required for CPPU alone, without changes to maximum core flow, [
] to allow for a comprehensive but more streamlined NRC staff review process.

A consequence of the simplified CPPU approach to evaluation of power uprate is the removal of some analyses normally included in the power uprate license amendment application to documents associated with core reload analyses. The reload analysis results are documented in the Supplemental Reload Licensing Report (SRLR), and the applicable core operating limits are documented in the plant- and cycle-specific Core Operating Limits Report (COLR). Although these documents are available for staff inspection, they are not submitted with the power uprate application and are not normally submitted for NRC staff review and approval.

For the CPPU LTR, it is assumed that the only change to the plant licensing and design basis is an increase of up to 20 percent over the 100 percent 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. Licensees proposing to reference this LTR as a basis for a power uprate license amendment request, and also, proposing to obtain a license amendment to incorporate one or more of the plant changes listed below must first request and obtain a license amendment for the associated change prior to the start of the staff review of the power uprate request that references this LTR.

- No increase in maximum normal operating reactor dome pressure

- No increase to maximum licensed core flow
- No increase to currently licensed MELLLA or MEOD upper boundaries

[ ]

The one exception is with regards to a source term methodology change. A licensee may submit and the NRC staff will review a source term methodology change, in lieu of the analysis in Section 9.2 of the CPPU LTR, concurrent with the power uprate request, if the source term submittal supports operation at the uprated power level. Licensees proposing to utilize fuel designs other than GE fuel, up through GE 14 fuel, may not reference the CPPU LTR as a basis for their power uprate since the CPPU LTR process applies only to GE fuel and GE accident analysis methods. However, such licensees may reference the CPPU LTR for areas other than those involving reactor systems and fuel issues which are not impacted by the fuel design.

The CPPU power/flow operating map is an extension of the current MELLLA or MEOD operating map. Therefore, the CPPU LTR and the safety evaluation are applicable only to plants that are currently licensed to operate with the MELLLA or the MEOD operational margin improvement option. A typical power/flow map showing the CPPU change in allowable operating conditions is shown on Figure 1-1 of the LTR.

The guidelines in ELTR1 and ELTR2 are also generally followed for CPPU evaluations. However, the CPPU safety analysis deviates from the ELTR1 and ELTR2 guidelines in the stability analyses and in the LOCA and transient analyses. Deviations and exceptions are noted in the following areas:

- Stability [ ]
- Emergency core cooling system (ECCS) performance [ ]
- Anticipated operating occurrences (AOO) [ ]
- Testing [ ]

The analyses and evaluations listed above would be performed in accordance with the proposed, more simplified CPPU approach. In the CPPU approach, [.]

Changes to the licensing and design basis necessary to support the licensing of power uprate at a plant will be reported and justified in a plant-specific power uprate licensing submittal. The plant-specific submittal will include and justify any proposed changes to the analysis basis methodology identified in ELTR1 and ELTR2, unless this methodology is revised by review and acceptance of the CPPU LTR. Applicable new methods that are reviewed and approved by the NRC independently of the CPPU LTR may be used after the approval is received and the methods are incorporated into the General Electric Standard Application for Reactor Fuel (GESTAR-II) (Reference 11). Any new methods that a licensee wishes to have reviewed or implemented 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.

Due to the effect of the CPPU process on simplifying many of the safety evaluations, a number of generic evaluations are cited by GENE to support the plant-specific submittals. In addition, some generic assessments from the ELTR1 and ELTR2 references can be utilized if they are shown to properly bound the effects from the CPPU approach. The CPPU LTR cites and summarizes results of these generic evaluations, generic assessments, and generic dispositions for NRC review and approval, and these may simplify the plant-specific NRC review required for future CPPU submittals.

To simplify future licensee amendment requests and to support NRC reviews of plant-specific CPPU submittals, the format of the PUSAR to be used for each plant-specific CPPU submittal will be based on the format of the CPPU LTR. The PUSAR is based on the above assumptions and includes consideration of the generic and plant-specific evaluations, assessments, and dispositions discussed in the CPPU LTR and the safety evaluation. Deviations from the generic bases and evaluations provided in the report will be included and justified in the plant-specific submittal. The level of information to be provided for each plant-specific submittal and the format for providing that information will still be consistent with past extended power uprate submittals. For those analyses and evaluations that are generically dispositioned in the CPPU LTR, the plant-specific PUSAR is required to list and to provide the basis for the generic dispositions and to confirm the applicability of these generic dispositions for the specific plant application. For any plant that seeks 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 GE 14 fuel, 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.

Limited technical specification (TS) setpoint changes are required as a result of the CPPU. Typically, setpoint changes are limited to the neutron monitoring system, main steamline high flow, and turbine first-stage pressure.

1.3 CPPU LTR Structure

The report section numbers and titles generally correspond to those used for previous plant-specific, extended power uprate submittals. Each of the evaluations included in those submittals were reviewed and assigned one of the two disposition categories:

- Generic assessment
- Plant-specific evaluation

Each primary section of the CPPU LTR begins with a table providing summary disposition for 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 judged to be potentially limiting with respect to safety analyses relative to the power uprate. Each principal evaluation is then 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 disposition category is included after the table. This justification may include current experience with extended power uprates and the basis for the disposition, as applicable.

The technical dispositions are contained in the CPPU LTR Sections 2 through 10. General information has also been provided in CPPU LTR 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 was not a focus of the NRC review. The utility may elect to reference some or all of the information given in Section 11 of the CPPU LTR in the documentation supporting the plant-specific licensing CPPU submittal.

1.3.1 Generic Assessments

Generic assessments are those safety evaluations that can be dispositioned for a group or for 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 the previous pressure increase power uprate assessments provided in ELTR-1 or ELTR-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 current GENE/GNF fuel designs up through GE 14, analyzed with standard NRC-approved GENE methodology. The effect of CPPU on future GENE/GNF fuel designs is to be addressed during the assessment of new fuel designs consistent with the requirements of the approved GESTAR-II. If another vendor fuel design is considered as part of the power uprate, fuel design dependent generic assessments will be separately evaluated and justified.

For those CPPU assessments judged to have a negligible effect, the current CPPU experience base plus a discussion of the evaluation of the event and justification for the assessment is provided in the CPPU LTR. ELTR-1 or ELTR-2 is referenced if the information in these reports supports the conclusion of a 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 safety evaluations affected by CPPU are fuel- and operating-cycle-(reload) dependent. Reload-dependent evaluations require that the reload fuel design, the reload core loading pattern, and the cycle-specific operating plan be established so that analyses can be performed to establish core operating limits. The core reload analysis demonstrates that the core design for CPPU meets the applicable NRC evaluation criteria and limits documented in GESTAR-II. Due to the lead time required for licensee power uprate submittals, the cycle-specific reload fuel design and the core loading pattern for the initial fuel cycle operation at uprated power are not established at the time of the plant-specific power uprate submittal.

Previous power uprate experience supports the GENE contention that the CPPU approach has a relatively small effect on certain cycle-specific reload analysis results affecting the core operating limits. Therefore, the reload fuel design and core loading pattern dependent plant evaluations for CPPU operation will be performed with the cycle-specific reload analysis as part of the GE standard reload licensing process. A plant cannot implement a power uprate unless the appropriate reload core analysis is performed and all criteria and limits documented in GESTAR-II are satisfied. Otherwise, the plant would be in an unanalyzed condition.

The proposed generic dispositions for reload analysis assessments are described in the appropriate sections of the staff's safety evaluation. For each of these assessments, an event 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 ELTR1 or ELTR2.

The applicability of the generic assessments for a specific plant application will be evaluated for each power uprate application. The plant-specific submittal will either document the confirmation of the generic assessment or will provide a plant-specific evaluation, consistent with Section 1.1 of the CPPU LTR, if the generic applicability assessment is unconfirmed. For any plant that seeks concurrent review or implementation of a power uprate and any of the excluded plant changes described in Section 1.0 of the CPPU LTR, 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.

1.3.2 Plant-Specific Evaluation

Plant specific evaluations are assessments of the principal evaluations that are not addressed by the generic assessments described in Section 1.3.1 of the CPPU LTR. The relative effect of the CPPU approach on plant-specific evaluations and methods used for their performance are provided in the CPPU LTR. Where applicable, the assessment methodology is referenced. If a specific approved computer code is used, the name of this computer code is provided in the subsection. If the computer code has been identified as approved in ELTR1, ELTR2, or GESTAR-II, the appropriate sections of these documents may be referenced instead of the original code reference.

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 justified in this safety evaluation.

For any plant that seeks 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.

1.3.3 Effect of CPPU

Operating Domain

The upper bound of the operating domain is defined by the current MELLLA/MEOD upper boundary. The current MELLLA/MEOD upper boundary remains unchanged with the CPPU approach in terms of absolute reactor power and reactor core flow, and the boundary is extended along the control rod line up to the uprated 10-percent core power value, as indicated on Figure 1-1 of the CPPU LTR. The effect of the CPPU on the other power flow map boundaries is provided in Table 1-2 of the CPPU LTR. Other changes in the plant operational flexibility or performance improvement options that affect the operating domain are not allowed, as noted in Section 1.0 of the CPPU LTR.

Nuclear and Thermal-Hydraulic Evaluations

An increase in the reactor power level affects the plant steady-state heat balance. The typical effect of a 20-percent increase in reactor power on plant operating parameters is shown in Table 1-3 of the CPPU LTR. 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 the fairly uniform effect of a 20-percent increase in power with no reactor pressure increase across the BWR fleet. A plant-specific power uprate submittal will include a summary of steady-state parameters based on the plant-specific CPPU heat balance.

The CPPU approach can also affect the thermal-hydraulic safety analyses. In ELTR-1 and ELTR-2, the effect of an EPU was generally shown to be limited. The CPPU approach has a smaller effect than an EPU with a pressure increase because of the constant pressure limitation. [

.] Some thermal-hydraulic safety analyses can be performed on a generic basis, as documented in the CPPU LTR. The remaining thermal-hydraulic safety analyses require plant-specific evaluations. The plant-specific evaluation or generic applicability confirmation will be provided in the plant-specific submittal.

The nuclear evaluation requirements and acceptance criteria for the limits will not be changed as a result of the CPPU approach. Specifically, the minimum cold shutdown margin and hot excess reactivity requirements identified in GESTAR-II remain applicable. [

.] The additional energy requirements for power uprate can be met by an increase in bundle enrichment and burnable poison loading, an increase in reload batch size, and/or changes in the fuel core loading pattern to achieve the required plant operating cycle length. The power distribution in the core is established to achieve the increased core power while satisfying the core operating limits. The required cycle specific nuclear analyses are performed as part of the standard reload analysis. The plant radiological analyses are more dependent upon power and are to be included in plant-specific submittals.

1.4 Regulatory Evaluation Summary

The staff evaluated the CPPU LTR for conformance with the generic BWR EPU program as defined in ELTR1 and ELTR2. The CPPU approach takes certain exceptions to certain previously approved generic positions in these topical reports. The staff reviewed the exceptions and the conclusions about their acceptability are given in the applicable sections of this evaluation.

The staff review of the CPPU approach used applicable rules, Regulatory Guides (RGs), Standard Review Plan (SRP) sections, and NRC staff positions on the topics being evaluated.

In the area of reactor core and fuel performance the staff applied: 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants" (or GDC); 10 CFR 50.46, "Acceptable criteria for emergency core cooling systems for light-water nuclear power plants;" SRP Sections 4.2, "Fuel System Design," and 4.3, "Nuclear Design."

In the area of reactor coolant system and connected systems the staff applied: 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements;" Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials;" 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation;" 10 CFR 50.36, "Technical specifications;" 10 CFR 50.55a, "Codes and standards;" Generic Letters (GL) 98-05, "Boiling Water Reactor Licensees Use of the BWRVIP-05 Report to Request Relief from Augmented Examination Requirements on Reactor Pressure Vessel Circumferential Shell Welds," GL 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," GL 95-07, "Pressure Locking and Thermal Binding of Safety Related Power Operated Gate Valves;" GL 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions;" Regulatory Guide 1.139, "Guidance for Residual Heat Removal Systems."

In the area of engineered safety features the staff applied: 10 CFR 50.46, "Acceptable criteria for emergency core cooling systems for light-water nuclear power plants;" 10 CFR Part 50, Appendix K, "ECCS Evaluation Models;" Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Post-Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants."

In the areas of instrumentation and control, electrical power and auxiliary systems, and power conversion systems the staff applied: Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Chapter 7; GDC 17, "Electric Power Systems;" 10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants."

In the area of radwaste systems and radiation sources the staff applied: 10 CFR 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low as Is Reasonably Achievable' for Radioactive Material in Light-water-cooled

Nuclear Power Reactor Effluents;" 10 CFR Part 20, "Standards for Protection Against Radiation;" GDC 19, "Control room;" 40 CFR Part 190.

In the area of reactor safety performance evaluation the staff applied: Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Chapter 15; GDC 10, "Reactor Design;" GDC 15, "Reactor Coolant System Design;" GDC 17, "Electric Power Systems;" GDC 20, "Protection system functions;" 10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants;" Regulatory Guide 1.155, "Station Blackout;" 10 CFR 50.63, "Loss of all alternating current power."

Regarding additional areas, including risk considerations and testing program, the staff applied: 10 CFR 50.49, "Environmental qualification of electrical equipment important to safety for nuclear power plants;" GDC 17, "Electric Power Systems;" RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis;" Regulatory Guide 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants."

The scope of the review for the CPPU approach included "lessons learned" from past power uprate amendment reviews. In reviewing the LTR, the staff considered the recommendations of the report of the Maine Yankee Lessons Learned Task Group (SECY-97-042, "Response to OIG Event Inquiry 96-04S Regarding Maine Yankee," February 18, 1997). The task group's main findings centered on the use and applicability of the computer codes and analytical methodologies used for power uprate evaluations.

1.5 Technical Evaluation Summary

The staff expects licensees requesting plant-specific CPPUs will identify all codes and methodologies used to obtain safety limits and operating limits and to explain how these limits were verified to be correct for the uprated core. Licensees will also be expected to identify and discuss any limitations imposed by the staff on the use of these codes and methodologies. A table should be provided, indicating that all the applicable codes were reviewed and approved by the NRC, with any exceptions being noted and individually justified. The licensee is expected to confirm having reviewed the results of GENE analyses to assure that the codes were used correctly by GENE for CPPU conditions and that the limitations and restrictions were followed appropriately by GENE. The licensee submittal, including the plant-specific PUSAR, may be used as the basis for an NRC audit of selected safety analyses and system and component performance evaluations used to support the power uprate.

As discussed earlier, plant-specific core reload analyses (i.e., those results documented in the SRLR and COLR) are not submitted with the licensee's power uprate application and are not normally submitted for NRC staff review and approval. The reload analyses are conducted using methods previously reviewed and accepted by the staff. Further, the methods approved for reload analyses specify the acceptance criteria for the transients to be analyzed. The reload evaluation process is documented in GESTAR-II. Existing regulations require licensees to obtain staff approval for changes to analysis methods and acceptance criteria used for reload analyses. In addition, based on previous experience with reviewing EPU analyses for EPUs that maintained a constant reactor dome pressure, the staff does not expect significant differences in the results of such analyses for pre- and post-CPPU conditions. Therefore, the staff determined that further review of the reload analysis methods or results was not necessary for CPPU applications. The staff may choose to audit certain future reload analyses for CPPU applicants.

The initial CPPU transition reload core will consist of fresh and exposed GNF GE-14 (10x10) fuel and also may have existing GE co-resident fuel of a different lattice design. The CPPU cycle-specific reload analyses and safety analyses will be performed in accordance with NRC-approved GE analytical methodologies described in the latest approved version of GESTAR-II. The licensing topical reports specifying the codes and methodologies used for performing the safety analyses are typically documented in Section 5 of the plant TSs. The limiting AOO and accident analyses are confirmed to be valid (or are re-analyzed) for every reload and the safety analyses of transients and accidents are documented in Chapter 15 of the plant updated final safety analysis report (UFSAR). Limiting transient or accident analyses are generally defined as analyses of events that could potentially affect the core operating and safety limits that ensure the safe operation of the plant. These reload analyses are not submitted for staff review and approval, but are available for staff audit. [

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1.6 Conclusions

The staff's review of the CPPU LTR found it acceptable for referencing to provide a consistent format and a guideline for future EPU applications that meet the limitations and restrictions noted in the CPPU LTR and further stated in this staff safety evaluation. The staff noted that the applicability of any generic assessment for CPPU will be confirmed by the licensee in the plant-specific licensing application or a plant/reload-specific evaluation that will be performed using approved methodology. The staff expects that reload-specific analyses will not be submitted with the power uprate application for NRC staff review and approval. Appropriate sections in this safety evaluation acknowledge those areas where analyses will be conducted for the core reload and may not be available during the staff's review of the power uprate application.

2.0 <u>REACTOR CORE AND FUEL PERFORMANCE</u>

This section of the CPPU LTR addresses the evaluations in RG 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Chapter 4, that are documented in the previous plant EPU submittals. The subsections reviewed are:

- Fuel Design and Operation
- Thermal Limit Assessment
- Reactivity Characteristics
- Stability
- Reactivity Control

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. [

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2.1 Fuel Design and Operation

For each fuel vendor, use of NRC-approved fuel design acceptance criteria and analysis methodologies assures that the fuel bundles perform in a manner that is consistent with the objectives of Sections 4.2 and 4.3 of the SRP and the applicable general design criteria (GDC) of Title 10 of the Code of Federal Regulations (10 CFR) Part 50, Appendix A.

Fuel bundles are designed to ensure that:

- the fuel bundles are not damaged during normal steady-state operation and AOOs;
- any damage to the fuel bundles will not be so severe as to prevent control rod insertion when required;
- the number of fuel rod failures during accidents is not underestimated; and
- the coolability of the core is always maintained.

The fuel vendors perform thermal-mechanical, thermal-hydraulic, neutronic, and material analyses to ensure that the fuel system design can meet the fuel design limits during steady-state, AOO, and accident conditions.

The effect of the CPPU approach on the fuel and core design and operation is described in the CPPU LTR. [.]

Fuel design limits are established [ ] for all new fuel product line designs as a part of the fuel introduction and reload analyses using the approved GESTAR-II process.

The power level above which fuel thermal margin monitoring is required may change with the implementation of the CPPU. The original plant operating licenses set this monitoring threshold at a typical value of 25 percent of rated thermal power. [.] For CPPU, the fuel thermal margin monitoring threshold is scaled down, if necessary, to ensure that the monitoring is initiated [.] A change in the fuel thermal monitoring threshold also requires a corresponding change to the TS reactor core safety limit for reduced pressure or low core flow.

The staff agrees that this [ ], the plant must perform a [ ] evaluation using an approved methodology. This approach is acceptable to the staff.

2.2 Thermal Limit Assessment

GDC 10 of 10 CFR Part 50, Appendix A, requires that the reactor core and the associated control and instrumentation systems be designed with appropriate margin to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operation, including AOOs. Operating limits are established to assure that regulatory and/or safety limits are not exceeded for a range of postulated events (transients and accidents).

The effect of the CPPU on the minimum critical power ratio (MCPR) safety and operating limits and on the maximum average planar linear heat generation rate (MAPLHGR) and linear heat generation rate (LHGR) limits is discussed in the CPPU LTR. The topics considered include:

- Safety Limit MCPR
- MCPR Operating Limit
- MAPLHGR Limit
- Maximum LHGR Limit

The safety limit minimum critical power ratio (SLMCPR) ensures that 99.9 percent of the fuel rods are protected from boiling transition during steady-state operation. The operating limit minimum critical power ratio (OLMCPR) assures that the SLMCPR will not be exceeded as the result of an AOO.

The MAPLHGR operating limit is based on the most limiting loss-of-coolant accident (LOCA) and ensures compliance with the ECCS acceptance criteria in 10 CFR 50.46. For every new fuel type, the fuel vendors perform LOCA analyses to confirm compliance with the LOCA acceptance criteria, and for every reload licensees confirm that the MAPLHGR operating limit for each reload fuel bundle design remains applicable.

In general, the licensee must ensure that plant operation is in compliance with the cycle-specific thermal limits (SLMCPR, OLMCPR, MAPLHGR, and maximum LHGR) and specify the thermal limits in a cycle-specific COLR as required by Section 5 of the plant TS. In addition, while uprated power operation may result in a small change in average fuel burnup, the licensee cannot exceed the NRC-approved maximum burnup limits. In accordance with Section 5 of the TS, any cycle-specific analyses are performed using NRC reviewed and approved methodologies. Therefore, the staff expects that the licensee will appropriately consider the potential effects of uprated power operation on the fuel design limits, and that the current thermal limits assessment will show that the plant can operate within the fuel design limits during steady-state operation, AOOs, and accident conditions.

The staff agrees that this [

], the plant must perform a [ ] evaluation using an approved methodology. This approach is acceptable to the staff.

2.3 Reactivity Characteristics

The effect of the CPPU approach on the minimum shutdown margin and hot excess reactivity is discussed in the CPPU LTR. The topics addressed in this evaluation are:

- Hot excess reactivity
- Shutdown margin

The higher core energy requirements of a power uprate may affect the hot excess core reactivity and can also affect operating shutdown margins. The general effect of a power uprate on core reactivity, as described in Section 5.7.1 of ELTR-1, is also applicable to a CPPU. Based on experience with previous plant-specific power uprate submittals, the required hot excess reactivity and shutdown margin can typically be achieved for power uprates through the standard approved fuel and core reload design process. Plant shutdown and reactivity margins must meet NRC-approved limits established in GESTAR-II on a cycle-specific basis and are evaluated for each plant reload core, [

].

The CPPU reload core design will account for any loss of margin for future cycles. The reload core analysis will ensure that the minimum shutdown margin requirements are met for each core design and that the current design and TS cold shutdown margin will be met. Since the licensee will continue to confirm that the TS cold shutdown requirements will be met for each reload core operation, the staff finds this acceptable.

The staff agrees that this [

], the plant must perform a [ ] evaluation using an approved methodology. This approach is acceptable to the staff.

2.4 Stability

The staff review in the area of reactor stability is conducted to ensure that the requirements of GDC 12 of 10 CFR Part 50, Appendix A, "Suppression of reactor power oscillations," are satisfied.

The CPPU LTR has taken exception to one of the generic guidelines in ELTR2, regarding thermal hydraulic stability. The staff SE on ELTR2, Section 3.2.2, "Long Term Solution," states: "The prevention and detection/suppression features of the long term stability solutions are either demonstrated to be unaffected by power uprate or are modified and validated in accordance with the solution methodology." The ELTR2 staff SE requires that the thermal hydraulic stability monitoring and monitoring system be validated in accordance with the generic solution methodology using a representative equilibrium core design and included in the application for EPU. [

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Section 3.2 of ELTR-2 documents interim corrective actions (ICA) and four long-term solution (LTS) stability options: Enhanced Option I-A, Option I-D, Option II, and Option III.

A generic evaluation was performed for the ICAs as documented in Section 3.2.1 of ELTR-2. This generic evaluation is applicable for the CPPU. Interim corrective action stability boundaries are the same in terms of absolute core power and flow. The listed power levels, as a percentage of rated power, are scaled [ ] based on the new uprated power.

For the long-term solution options, evaluations are reload core dependent and are performed for each reload fuel cycle. The analyses of each long-term option are addressed in the CPPU LTR. The topics addressed in this evaluation are:

- Enhanced Option I-A
- Option I-D

- Option II
- Option III (OPRM armed region and trip/Hot channel oscillation magnitude)

2.4.1 Plants with Enhanced Option I-A

The stability regions and associated trip setpoints may change with CPPU. Enhanced Option I-A (E1A) is classified as 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. These features are either confirmed or adjusted for each plant reload. The trip function settings and monitored region for the CPPU will be established by the [ ] analysis that incorporates the uprated power level.

The staff agrees that this [

], the plant must perform a [ ] evaluation using an approved methodology. This approach is acceptable to the staff.

2.4.2 Plants with Option I-D

The exclusion region may change and SLMCPR protection may be affected by the CPPU. 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 that the existing flow-biased flux trip provides adequate SLMCPR protection for events that initiate along the rated rod line. These features will be analyzed for the [ ] analysis that incorporates 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 [ ] analysis that incorporates the new rated power level.

The staff agrees that this [

], the plant must perform a [ ] evaluation using an approved methodology. This approach is acceptable to the staff.

2.4.3 Plants with Option II

The exclusion region may change and SLMCPR protection may be affected by CPPU. 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. These features will be analyzed for the [ ] analysis that incorporates the uprated power level.

The staff agrees that this [

], the plant must perform a [ ] evaluation using an approved methodology. This approach is acceptable to the staff.

2.4.4 <u>Plants with Option III</u>

The Option III trip setpoint may be affected by CPPU operating conditions. The OPRM armed region will be rescaled with CPPU. Option III is a detect-and-suppress solution, which combines closely spaced LPRM detectors into "cells" to effectively detect any mode of reactor instability. Evaluation is dependent upon the core and fuel design and is performed for each reload. The generic analyses for the Option III hot channel oscillation magnitude and the OPRM hardware were designed to be independent of core power. [

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The staff agrees that this [

], the plant must perform a [ ] evaluation using an approved methodology. This approach is acceptable to the staff.

2.5 Reactivity Control

The control rod drive (CRD) system controls gross changes in core reactivity by positioning neutron-absorbing control rods within the reactor. The CRD system is also required to scram the reactor by rapidly inserting withdrawn rods into the core. The scram, rod insertion and withdrawal functions of the CRD system depend on the operating reactor pressure and the pressure difference between the CRD system hydraulic control unit (HCU) and the reactor vessel bottom head pressure. The CRD system was generically evaluated in Section 5.6.3 and J.2.3.3 of ELTR1 and in Section 4.4 of Supplement 1 to ELTR2. The [ ] evaluation concluded that the CRD systems for BWR/2-6 plants are acceptable for EPU as high as 20 percent above the original rated power. Therefore, no additional plant-specific calculations are required beyond confirmatory evaluation.

The topics considered in this section are:

- Scram Time Response (BWR/6 and BWR/2-5)
- CRD Positioning
- CRD Integrity

2.5.1 <u>Control Rod Scram</u>

In pre-BWR/6 plants, the scram times may be decreased by the transient pressure response. [.] The pre-BWR/6 plant generic scram times for American Society of Mechanical Engineers (ASME) overpressure protection and critical power ratio pressurization transient analyses may not be adversely affected by the reactor transient pressure. Thus, the analyses and results would remain valid.

For BWR/6 plants, the increase in the transient pressure response tends to increase the scram time. Because the normal steady-state reactor dome pressure for the CPPU does not change, the scram time performance relative to pre-power uprate plant operation may [.] The BWR/6 design generic scram times for ASME overpressure protection and AOO analyses are based on generic reactor pressure versus time envelopes. The overpressure evaluation described in Section 3.1 of the CPPU LTR will be used to confirm that the transient reactor pressures remain within the generic envelopes.

In addition, scram time testing verifies the scram time for individual control rods. [.] The staff SE for ELTR2 states that "the plant-specific submittal for BWR/6 plants must provide assurance that the scram insertion speeds used in the transient analyses are slower than the requirements contained in the plant."

The staff agrees that this [ ], the plant must perform a [ ] evaluation using an approved methodology. This approach is acceptable to the staff.

2.5.2 <u>Control Rod Drive Positioning</u>

The increase in reactor power at the CPPU operating condition results in [.] The automatic operation of the system flow control valve maintains the required drive water pressure [.] The normal CRD positioning function is an operational consideration and is not a safety-related function.

2.5.3 <u>Control Rod Drive Integrity Assessment</u>

GENE indicated that 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. [.] In its response to the staff's RAI dated December 18, 2001, (Reference 4), GENE indicated that in those cases where the existing design basis conditions do not bound CPPU conditions, a plant-specific evaluation of the CRD mechanism will be performed to account for other applicable design basis mechanical loads resulting from the reactor vessel motion.

On the basis of its review, the staff agrees with GENE's approach that confirmation of bounding existing design basis or plant-specific evaluations accounting for design basis mechanical loads affecting CRDMs would provide the basis to ensure that the CRDMs meet design basis and performance requirements at CPPU conditions.

3.0 REACTOR COOLANT AND CONNECTED SYSTEMS

The staff's review of the CPPU LTR focused on areas with regard to the structural and pressure boundary integrity of the piping systems and components, their supports, and reactor vessel and internal components and the control rod drive mechanisms (CRDM), and the balance-of-plant (BOP) piping systems.

The previous GENE generic topical reports on guidelines and evaluation for 120 percent BWR extended power uprate, known as ELTR1 and ELTR2 respectively, were based on a 24 percent higher steam flow; an operating temperature increase of approximately 10°F; and an operating pressure increase up to approximately 75 psi. The CPPU approach assumes that the maximum reactor vessel dome pressure remains unchanged from the licensed power level, and the dome temperature is also unchanged. The steam flow rate will increase up to approximately 24 percent, similar to that specified in ELTR1 and ELTR2. The maximum core flow rate remains unchanged for the CPPU.

3.1 Nuclear System Pressure Relief/Overpressure Protection

The design pressure of the reactor vessel and reactor coolant pressure boundary (RCPB) remains at 1250 psig. The ASME Code allowable peak pressure for the reactor vessel and the RCPB is 1375 psig (110 percent of the design pressure of 1250 psig), which is the acceptance limit for pressurization events. [

.]

Section 5.5.1.4 and Appendix E of ELTR1 evaluated the ASME overpressure analysis in support of a 20-percent power increase, stating that the limiting pressurization transient events are the main steam isolation valve (MSIV) closure and turbine trip with turbine bypass failure (TTNBP). However, the MSIV closure has been determined generically to be the more limiting event. The staff-approved evaluation model ODYN is used for the CPPU overpressure protection analysis and this is consistent with the generic analysis in Section 3.8 of ELTR2.

[

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The safety relief valves (SRVs) provide overpressure protection for the NSSS, preventing failure of the nuclear system pressure boundary and uncontrolled release of fission products. In general, the SRVs are piped to the suppression pool. These SRVs, together with the reactor scram function, provide overpressure protection. The SRV setpoints are established to provide the overpressure protection function while ensuring that there is adequate pressure difference (simmer margin) between the reactor operating pressure and the SRV actuation setpoints. The SRV setpoints are also selected to be high enough to prevent unnecessary SRV actuations during normal plant maneuvers.

The CPPU evaluation is generally consistent with the generic evaluations and discussions in Section 5.6.8 of ELTR1 and Section 3.8 of ELTR2.

The limiting ASME code overpressure analyses (discussed in Section 3.1 of the LTR) is based on 102 percent of the CPPU power level. The current SRV setpoints and upper tolerance limits will not generally change. The ASME overpressure situation is evaluated [.] Therefore, the capability of the SRVs to ensure ASME overpressure protection will be confirmed [.]

3.2 Reactor Pressure Vessel (RPV) and Internals

The RPV structure and support components form a pressure boundary to contain the reactor coolant and moderator, and form a boundary against leakage of radioactive materials into the drywell. The RPV also provides structural support for the reactor core and internals. Many reactor vessel components are not significantly impacted by CPPU. [,] the plant-specific evaluation will be performed consistent with the methods documented in Appendix I of ELTR1. Plant-specific evaluations will report the maximum stresses and fatigue usage factors for the limiting reactor vessel components and supports, that are required to meet the allowable limits in accordance with the existing design basis.

The CPPU LTR evaluated the reactor vessel internal components in accordance with their current design basis. The loads and load combination 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. [.] The evaluation for CPPU includes internal components such as shroud, shroud support, core plate, top guide, jet pumps, fuel channel, orificed fuel support, control rod guide tube, control rod drive housing, control rod drive mechanism, shroud head and separators, access hole cover, feedwater (FW) sparger, core spray line and sparger steam dryer, and low pressure core injection (LPCI) coupling. 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. For components where the CPPU conditions are bounded by the design basis analyses, no further evaluation is performed. For other components, the plant-specific evaluations will be performed consistent with the methods documented in Appendix I of ELTR1. Plant-specific evaluations will report the maximum stresses and fatigue usage factors for the limiting reactor internal components against the acceptable limits 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.

On the basis of its review, the staff finds the proposed CPPU methodology acceptable. The staff concludes that the performance of the [ ] evaluations at CPPU conditions should provide the basis to determine the acceptability of stresses and fatigue usage factors of the limiting reactor vessel and internal components when compared against allowable code limits.

3.2.1 <u>Reactor Vessel Fracture Toughness</u>

The CPPU LTR stated that CPPU may result in a higher operating neutron flux at the reactor pressure vessel wall, consequently increasing the integrated flux over time (neutron fluence). The report stated that any licensee seeking future CPPU will need to perform a plant-specific vessel wall neutron fluence analysis consistent with NRC-approved methods. The licensee will also need to assess the effect of the change in neutron fluence on the adjusted reference temperatures (ART) values and upper shelf energy (USE) values for the RPV materials. Further, any increase in ART values and decrease in USE values for a given material will be calculated in accordance with RG 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2. With regard to evaluating the effect of CPPU on the RPV ART values and pressure-temperature (P-T) limits, GENE stated that, for the case where the plant's P-T limit curves are limited by the most limiting ART for the RPV beltline materials, the increase in the ART will also require a revision to the P-T limit curves. The new P-T limit curves are to be based on meeting the requirements related to P-T limit curves in 10 CFR Part 50, Appendix G. Those requirements provide adequate margins of safety during normal operations, including anticipated operational transients and systems hydrostatic tests, to which the pressure boundary may be subjected over its service life.

With regard to evaluating the effect of the CPPU on USE, GENE stated that the USE values for the vessel materials at end of life must remain above 50 ft-lb. The criteria cited by GENE is consistent with the requirements of 10 CFR Part 50, Appendix G. If a USE value for a given RPV material does not meet the 50 ft-lb criterion, or if the available data are insufficient to determine what the USE value is, an equivalent margins analysis (EMA) can be performed to demonstrate that lower values of USE will provide acceptable margins of safety for the RPV material. In the report, GENE stated that it had performed a generic EMA for the RPV materials of the U.S. BWR fleet in Reference 12, which was approved by the NRC in an SE to Gulf States Utilities Company dated December 8, 1993. Although the generic EMA indicated that all BWR RPV materials would meet the EMA requirements, [

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The staff concurs that applicants for the power uprate will need to perform revised plant-specific neutron fluence assessments for the RPV materials and that those assessments must be performed in accordance with an NRC-approved methodology. The plant-specific assessments for calculating the P-T limits and USE will be based on these neutron fluence assessments and will need to comply with the following:

- Section 50.60(a) requires that plants meet the fracture toughness and material surveillance program requirements for the RCPB specified in Appendices G and H to 10 CFR Part 50. Section 50.60(b) specifies that proposed alternatives to the described requirements of 10 CFR Part 50, Appendices G and H, may be used when an exemption is granted by the Commission under the provisions of 10 CFR 50.12.

- Section 50.36 requires that the P-T limits for a given facility be included as part of the limiting conditions for operation in the plant TS. Proposed changes to the P-T limits therefore need to be submitted as license amendment requests pursuant to the requirements of 10 CFR 50.90. Section IV.A.2. of 10 CFR Part 50, Appendix G provides the criteria for generating these P-T limits. With regard to generation of the P-T limits, 10 CFR Part 50, Appendix G requires that the P-T limits must be at least as conservative as those that would be generated if the methods of analysis in Appendix G to Section XI of the ASME Code were used to generate the curves. Section IV.A.2. of 10 CFR Part 50, Appendix G, also specifies minimum temperature requirements for the operation of the reactor. Pursuant to 10 CFR 50.60(b), licensees must request appropriate exemptions from the requirements of Section IV.A.2. of 10 CFR Part 50,

Appendix G, if their P-T limit curves are less conservative than those that would be generated if the methods of Appendix G to Section XI of the ASME Code were used to generate the curves, or if the P-T limits do not satisfy the applicable minimum temperature requirements for RPV specified in 10 CFR Part 50, Appendix G. The NRC will evaluate all exemption requests on a case-by-case basis against the exemption acceptance criteria of 10 CFR 50.12.

- Section III.A. to 10 CFR Part 50, Appendix G, requires that ferritic materials of the RPV beltline and other regions of the reactor coolant pressure boundary be evaluated in accordance with the requirements of the ASME Code and that surveillance materials must be tested in accordance with the requirements specified in 10 CFR Part 50, Appendix H. Paragraph F of Appendix G of 10 CFR Part 50, defines the beltline as follows: "Beltline or Beltline region of reactor vessel means the region of the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active fuel core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection for the most limiting material with regard to radiation damage." The threshold for monitoring used by the NRC is listed in 10 CFR Part 50, Appendix H, as $1X10^{17}$ n/cm2 (E 1.0 MEV). Applicants for the CPPU will need to perform a plant-specific evaluation of the effect of the CPPU on the end-of-life neutron fluence levels (E 1 MEV) for the RPV materials to determine (1) whether additional materials need to be added to those previously listed for the beltline of the RPV, and (2) what the change is to the fluence levels for the materials listed for the beltine region of the vessel. The applicant must evaluate each non-beltline material expected to experience an end-of-life fluence greater than $1X10^{17}$ n/cm2. Those materials that have the potential to become limiting will be added to the materials listed for the beltline region of the vessel.

Applicants seeking to use the LTR as their basis for CPPU license amendments will have to evaluate all beltline materials for ART and USE based on the CPPU-based fluence values. The ART is evaluated for beltline materials including any materials that are added to the beltline list. The current plant specific PT limit curves are evaluated relative to the change in ART. If the change in ART results in new and bounding PT limit curves, GENE will recommend that the PT curves be revised. Pursuant to 10 CFR 50.90, if this occurs, the applicant must submit a license amendment request for NRC approval of the new limiting PT curves.

Licensees seeking to use the LTR as their basis for CPPU license amendments must assess the effects of the CPPU-based fluence levels on the acceptability of the USE values for the RPV beltline materials. The licensee must demonstrate that either the USE values for all beltline materials, as determined from the CPPU-based fluence levels, will remain above 50 ft-lb throughout the licensed life of the plant, or that GENE's staff-approved generic EMA analysis, as provided in Reference 12 remains bounding for their CPPU-based USE values. If a licensee cannot satisfy these conditions, the licensee must submit a revised, plant-specific EMA analysis for its RPV beltline materials demonstrating compliance with Section IV.A.1 of 10 CFR Part 50, Appendix G. This is consistent with Section 3.2.1 of Revision 1 to the LTR.

The staff has noted that Revision 1 to the LTR does not address the potential effects of neutron fluences resulting from power uprate on relief requests that have been previously submitted as alternative programs to the augmented inspection requirements of 10 CFR 50.55a(g)(6)(ii)(A), and that have been approved in conformance with the acceptable alternative program provisions of 10 CFR 50.55a(a)(3). Applicants for CPPU that have obtained approvals for relief from the augmented inspection requirements specified in 10 CFR 50.55a(g)(6)(ii)(A) must evaluate the effect of the neutron fluences resulting from power uprate on their probabilistic fracture mechanics and ART assessments that were previously performed (i.e., Generic Letter (GL) 98-05, "Boiling Water Reactor Licensees Use of the BWRVIP-05 Report to Request Relief from Augmented Examination Requirements on Reactor Pressure Vessel Circumferential Shell Welds," assessments) to determine if the previous assessments remain bounding and valid. Licensees must verify that their technical basis for continued relief from the requirements of 10 CFR 50.55a(g)(6)(ii)(A) remains valid or revise their RPV inspection program to comply with the regulations.

On the basis of the above review, the staff concludes that demonstration of the performance of the reactor vessel materials will be dependent on plant-specific evaluations under CPPU conditions using plant-specific design and as-built information.

3.3 Flow-Induced Vibration

The flow-induced vibration (FIV) levels will increase in proportion to the increase in the fluid density and the square of the fluid velocity following the proposed power uprate. The FIV evaluation addresses the influence of an increase in flow during CPPU on RCPB piping, RCPB piping components and RPV internals.

The CPPU LTR evaluated the FIV effect on the RPV internal components using the recorded FIV data of all instrumented internal components for extrapolation of the CPPU vibration levels. Components in the [ ] are not affected by the CPPU since the core flow remains unchanged. Components in the [ ] that are affected by FIV due to the increase in feedwater, recirculation drive and steam flow will be evaluated [.] Components such as jet pump assemblies, jet pump sensing lines, feedwater sparger and steam separators are evaluated at the maximum core flow point for the uprated power level, based on available vibration data from the specific plant and/or from another plant of the same or similar design. The plant-specific evaluation includes assessment of plant startup data, dynamic structural analysis and, if necessary, fatigue usage determination. The staff concurs with the CPPU LTR regarding the performance of plant-specific FIV evaluation, and if necessary, fatigue determination for components affected by an increase in flow during CPPU.

CPPU will increase main steam and feedwater flow approximately 24 percent. The flow-induced vibration levels are expected to increase by approximately [.] To ensure that the vibration level will be below the acceptable limit, a startup vibration assessment will be required during the initial implementation of CPPU. The startup testing would include monitoring and evaluating flow-induced vibration during initial plant operation at CPPU conditions. The remote vibration monitoring sensors will be used for piping inside the containment and for areas that are inaccessible to plant personnel when the plant is at high power levels. The vibration testing will involve the performance of visual observations and conducting vibration measurements using hand-held vibration instruments during walkdown for piping outside the containment. In its response dated December 18, 2001, (Reference 4) to the staff's RAI, GENE provided the test procedure, analysis and acceptance criteria of the startup vibration test to be conducted on a plant-specific basis. The staff finds the vibration test methodology and acceptance criteria consistent with ASME Section III and ANSI/ASME

OM-S/G-1997 Code, "Requirements for Pre-Operational and Initial Startup Vibration Testing of Nuclear Power Plant Piping," and therefore, acceptable.

3.4 Piping Systems and Components

The piping evaluation addresses the effects of CPPU due to increased flow rate, temperature and pressure on the RCPB and the BOP piping systems and components. The components evaluated included equipment nozzles, anchors, guides, penetrations, pumps, valves, flange connections, and pipe supports (including snubbers, hangers, and struts). The RCPB piping systems consist of safety-related piping subsystems that move fluid through the reactor and other safety systems. The BOP piping systems consist of piping subsystems that move fluid through systems that are not evaluated in conjunction with the RCPB piping systems.

The RCPB piping evaluations compare the changes in the design parameters such as flow, pressure, temperature, and mechanical loads between the current existing design basis and the CPPU conditions. For most RCPB piping systems such as the [ ], these design parameters will not increase. Consequently, there will be no change in pipe stress, pipe support loads (snubbers, hangers), and fatigue evaluations. For other safety-related piping systems such as the main steam, feedwater piping and associated branch piping as well as safety-related thermowells that are significantly affected by CPPU, an increase in the flow, pressure, temperature and mechanical loads will be evaluated [ ] consistent with the methods specified in Appendix K of ELTR1. Plant-specific evaluations are required to demonstrate that the calculated stresses and fatigue usage factors are less than the code allowable limits in accordance with the requirements of the applicable code of record in the existing design basis stress report. As such, the staff concludes that, where required, plant-specific analysis for CPPU would provide the basis to ensure that the RCPB piping systems and supports will continue to meet the code requirement and maintain the structural and pressure boundary integrity at the CPPU condition.

The evaluation of the BOP piping and appropriate components, connections and supports will be performed in a manner similar to the evaluation of the RCPB piping systems and supports. Results of the evaluation will be compared to the allowable limits in the original code of record such as ASME Code Section III. No new assumptions were introduced that were not included in the original analyses. In cases where the Code allowable values are not satisfied, detailed analyses or field modifications can be completed such that Code requirements are met. Pipe break locations and pipe whip restraint hardware capacities are also evaluated to demonstrate acceptability. The existing design analyses of the affected BOP piping systems were assessed on a plant-specific basis 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 allowable values, and analytical techniques will be used. The plant-specific evaluations will be performed to demonstrate that the calculated stresses and fatigue usage factors are less than the allowable limits in accordance with the requirements of the applicable code of record in the existing design basis stress report. As such, the staff concludes that the plant-specific analysis for the BOP piping systems would provide the basis to ensure that BOP piping will continue to maintain its structural and pressure boundary integrity at the CPPU condition.

A [ ] evaluation will be performed to address the effects of CPPU on the capacity and performance of safety and relief valves, air-operated-valves, motor-operated-valves and other safety-related valves. In its response to the staff RAI, dated December 18, 2001

(Reference 4), GENE required the plant-specific assessment to include consideration of GL 95-07, "Pressure Locking and Thermal Binding of Safety Related Power Operated Gate Valves," in addition to the assessment items associated with the evaluation of the containment system. Other evaluations include effects of CPPU on the plant-specific response and commitments to GL 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," for the plant MOV program and GL 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," for the overpressurization of penetration piping segments. The staff agrees with the CPPU LTR requirement to perform plant-specific evaluations relating to GL 89-10, GL 95-07 and GL 96-06 in consideration of CPPU.

On the basis of the above review, the staff concludes that although the method for the evaluation is consistent with Appendix K of ELTR1, the adequacy of affected piping, piping components, and their supports will be dependent on the plant-specific design and as-built information to demonstrate the structural and pressure boundary integrity of the RCPB and BOP piping systems and supports for the CPPU condition.

3.5 Main Steam Flow Restrictors

At normal operation, the main steam flow restrictors are required to pass a higher CPPU flow rate, which will result in an increased pressure drop. For the faulted condition with 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. Because the maximum operating dome pressure does not change, [.] Therefore, the main steam flow restrictors [.] Because the flow restrictors were designed and analyzed for the choked flow condition with the maximum pressure difference, which is bounding for the CPPU condition, the CPPU LTR concludes that the structural integrity of flow restrictors [.] The staff agrees with this conclusion.

Because the maximum operating pressure of the reactor steam dome will not change, the maximum flow rate through the flow restrictor is unchanged from the current analysis. Therefore, values from the current analysis for steam line break flow remain valid for uprate conditions.

3.6 Reactor Recirculation System

The primary function of the recirculation system is to vary the core flow and power during normal operation. However, the recirculation system also forms part of the reactor coolant system (RCS) pressure boundary.

The plant licensee must evaluate the changes in the system operating pressure and temperature at the CPPU conditions to either confirm that changes are small and result in conditions that remain within the current rated conditions, or to reevaluate. The CPPU will not result in an increase in the steady-state dome pressure. However, operation at the CPPU power level may increase the two-phase core flow resistance, requiring a slight increase in the recirculation system drive flow. The required pump head and pump flow at the CPPU conditions may increase the power demand of the recirculation motors slightly. CPPU does not generally require changes to the recirculation flow control system. The recirculation system evaluations should be consistent with the generic evaluation in ELTR2. Section 4.5 of

Supplement 1 to ELTR2 evaluated the recirculation system performance for a 20-percent power uprate with a 75 psig increase in the normal dome operating pressure and concluded that the recirculation system design can accommodate the operating condition associated with the power uprate.

The licensee should also confirm that CPPU conditions would not significantly increase the net positive suction head (NPSH) required or reduce the NPSH margin for the recirculation pumps.

In support of CPPU for a plant-specific application, the licensee will also re-analyze the anticipated transient without scram (ATWS) with recirculation pump trip. Evaluation of the ATWS is discussed in Section 9.3.1 of the CPPU LTR. The staff concludes that the impacts on the recirculation system safety functions discussed in Supplement 1 to ELTR2 will be adequately considered for CPPU applications.

3.7 Main Steam Isolation Valves (MSIV)

The MSIVs are part of the RCPB and perform a safety function (i.e., steam line isolation). The MSIVs must be able to close within the specified time limits at all design and operating conditions upon receipt of a closure signal. They are designed to satisfy leakage limits set forth in the plant TS.

The MSIVs were generically evaluated in Section 4.7 of Supplement 1 to ELTR2. The generic evaluation covered the effects of the power uprate changes on: (a) the capability of the MSIVs to meet pressure boundary structural requirements, and (b) the safety function of the MSIVs.

CPPU conditions are typically bounded by the conditions assumed [.] The increased steam flow will assist in the closure of the MSIVs. The licensee is expected to adjust the actual in-plant closure rate so that the MSIV closure time will be maintained within the required TS range.

The staff accepts the [ ] assessment that the MSIV closure time can be maintained as analyzed and specified in the TS. In addition, various TS surveillances require routine monitoring of MSIV closure time and leakage to ensure that the licensing basis for the MSIVs is preserved.

Based on the review of the CPPU evaluation and rationale, the staff agrees with the conclusion that CPPU operation, as indicated above, [ ] and that the plant operations at the EPU level will not affect the ability of the MSIVs to perform their safety function.

3.8 Reactor Core Isolation Cooling (RCIC)/Isolation Condenser

The RCIC system provides core cooling in the event of a transient where the RPV is isolated from the main condenser, concurrent with the loss of all feedwater flow (LOFWF), and when the RPV pressure is greater than the maximum allowable for the initiation of a low-pressure core cooling system.

Section 5.6.7 of ELTR1 provides the scope of the RCIC system evaluation. The maximum injection pressure for RCIC is conservatively based on the upper analytical setpoint for the lowest set available group of SRVs operating in the relief mode. For the CPPU condition, the reactor dome pressure does not change, and the SRV setpoints should remain unchanged, and there would be no changes to the RCIC high pressure injection parameters. The RCIC injection rate required at CPPU conditions should also be unchanged from the system design flow rate. The RCIC turbine operation at CPPU should not change any startup transient or affect system reliability. Either the RCIC system has been modified to include the startup control function concept presented in GENE guidance in Reference 13 or the licensee will provide an evaluation and justification for no modification. [

.] The required CPPU surveillance testing and system injection demands would occur at the same reactor operating pressures, so there would be no change to existing system and component reliability. The loss of feedwater (LOFW) transient event will be evaluated [ ], and the acceptance criterion, (to maintain reactor water level above top of active fuel) will continue to be met for CPPU conditions.

Because the licensee will analyze the LOFW transient for CPPU operation, consistent with the ELTR1 guidelines, and will conservatively evaluate the pressure performance requirements of the RCIC system, the staff accepts the CPPU LTR assessment.

3.9 Residual Heat Removal (RHR) System

The RHR system is designed to restore and maintain the reactor coolant inventory and to remove sensible and decay heat from the primary system and containment following reactor shutdown for both normal shutdown and post accident conditions. The RHR system is designed to operate in the low-pressure coolant injection (LPCI) mode, the suppression pool cooling (SPC) and containment spray cooling (CSC) modes, the shutdown cooling (SDC) mode, the steam condensing mode, and the fuel pool cooling assist mode. The LPCI mode is discussed in Section 4.2.4 of the LTR. The SPC and CSC modes are addressed in Section 4.1 of the CPPU LTR. The fuel pool cooling assist mode is discussed in Section 6.3.1 of the CPPU LTR. The effects of the CPPU on the RHR modes are described below. The results of the following evaluations are consistent with the [ ] evaluation in Section 4.1 of ELTR2.

During normal plant operation, the suppression pool cooling function is to maintain the suppression pool temperature below the TS limit. Following abnormal events, the SPC function controls the long-term suppression pool temperature such that the design temperature limit is not exceeded. The proposed CPPU would increase the reactor decay heat, which increases the heat input to the suppression pool during a LOCA, and results in a higher peak suppression pool temperature. The effect of CPPU on the suppression pool after a design basis LOCA is discussed in Section 4.1 of the CPPU LTR. In Section 4.1 of this staff evaluation, the staff accepted the proposed approach for CPPU to conduct [ ] reviews of the containment effects of a LOCA.

The containment spray cooling mode provides suppression pool water to the spray headers in the containment to reduce containment pressure and temperature during post-accident conditions. [

.] The effect of the containment spray on containment is discussed in Section 4.1 of the CPPU LTR.

The operational objective for a normal shutdown is to reduce the bulk reactor temperature after scram to an acceptable value (typically 125°F) within a specified time (approximately 20 hours) using the available SDC heat exchanger loops. A single loop cooldown target may be specified (e.g., to reach 212°F within 20 hours). The staff accepts the proposed approach that licensees will conduct plant-specific SDC evaluations at the CPPU condition to demonstrate that plants can meet the required cooldown time.

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. If needed, the assist mode of the RHR system uses the RHR heat removal capacity to provide supplemental fuel pool cooling in the event that the fuel pool heat load exceeds the heat removal capacity of the fuel pool cooling and cleanup system. This mode can be operated separately or along with the fuel pool cooling and cleanup system to maintain the fuel pool temperature within acceptable limits. Operation of this mode is discussed in Section 6.3.1 of the LTR. The CPPU LTR requires a plant-specific evaluation of the impact of operations at the CPPU conditions on the assist mode of the RHR system.

Based on its review of the CPPU evaluation and rationale, the staff concludes that plant operation at the proposed CPPU conditions would be expected to have acceptable impacts on the different modes of the RHR system. This is based on the proposed approach that relies on analysis methods previously accepted by the NRC in ELTR2. Licensees are expected to provide confirmation of the impact of operation at CPPU conditions on RHR system capabilities in plant-specific submittals.

3.10 Reactor Water Cleanup (RWCU) System

The RWCU is designed to remove solids and dissolved impurities from the reactor coolant, thereby reducing the concentration of radioactive and corrosive species. The RWCU is a normally operating system with no safety-related functions other than containment isolation. The increase in feedwater flow after power uprate will slightly affect its operation. Higher feedwater flow increases the input of contaminants to the reactor and results in a slight increase in the normal level of conductivity and activated corrosion products. Also, increase in the feedwater line pressure will have slight effect on the system operating conditions related to the containment isolation. This function of the RWCU is addressed in Section 4.1 of the report in the containment system performance evaluation. [

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The staff reviewed the methodology described in the CPPU LTR for evaluating the effects of power uprate on the performance of the RWCU. The methodology [

] evaluations of the effects of power uprate and to report the results of these evaluations. The staff finds this method of evaluation of the effects of power uprate on the performance of the RWCU acceptable.

3.11 Conclusion

To support the proposed CPPU approach, GENE has provided analytical evaluations of affected plant components and equipment, and required evaluations of existing generic communications applicable to the extended power uprate (up to 20-percent increase in core thermal power). Based on its review, the staff has determined that CPPU LTR in conjunction with the generic guidelines and evaluation provided in ELTR1/ELTR2 coupled with a plant-specific licensing evaluation, would provide the information necessary for the staff's review of individual applications for extended power uprate in the areas of structural and pressure boundary integrity for piping systems, components, and their supports, reactor vessel and internals, core support structure, and the control rod drive mechanisms.

4.0 ENGINEERED SAFETY FEATURES

The staff evaluated the system/program performance of the following structures, systems, components, and regulatory programs, as well as anticipated operational occurrences, and design basis accidents described in the CPPU LTR:

- Steam Dryer/Separator Performance
- Main Steamline Flow Restrictions
- Main Steam Isolation Valves
- Residual Heat Removal System (including Suppression Pool Cooling, Containment Spray Cooling, and Fuel Pool Cooling Assist Modes)
- Containment System Performance (including Pressure and Temperature Response, Containment Dynamic Loads, Containment Isolation, and Generic Letter 96-06)
- ECCS Net Positive Suction Head
- Main Control Room Atmosphere Control System
- Standby Gas Treatment System
- Post-LOCA Combustible Gas Control System
- Fuel Pool Cooling
- Water Systems
- Power Dependent Heating, Ventilation, and Air Conditioning
- Fire Protection Program
- Turbine-Generator
- Condenser and Steam Jet Air Ejectors
- Turbine Steam Bypass
- Feedwater and Condensate Systems
- Liquid Waste Management
- Gaseous Waste Management
- Station Blackout
- High Energy Line Breaks
- Moderate Energy Line Breaks
- Environmental Qualification of Mechanical Equipment with Non-Metallic Components
- Mechanical Component Design Qualification

Plant-Specific Evaluations

The CPPU LTR identified the following structures, systems, components, regulatory programs, anticipated operational occurrences, and design basis accidents as subject to plant-specific evaluation without defining a specific method:

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Because the LTR does not constrain the scope of reviews in these areas, the plant-specific approach regarding system/program performance is acceptable without further evaluation.

4.1 Containment System Performance

The CPPU LTR states that the containment evaluation will be based on the methodology described in Section 5.10.2 of ELTR1. This approach involves a plant-specific analysis of the effect of the uprate power on containment. These evaluations will include containment pressures and temperatures, LOCA containment dynamic loads, safety-relief valve containment dynamic loads and subcompartment pressurization.

Appendix G of ELTR1 prescribes the generic approach for this evaluation and outlines the methods and scope of plant-specific containment analyses to be done in support of power uprate. These analyses will cover the response through the time of peak drywell pressure throughout the range of power/flow operating conditions with power uprate. Appendix G states that the licensee will analyze short term containment pressure and temperature response using the previously applied GE M3CPT code.

The approach also includes an evaluation of long-term containment heatup (i.e., suppression pool temperature) for the limiting safety analysis report events to show that pool temperatures will remain within limits for suppression pool design temperature, ECCS NPSH, and equipment qualification temperatures. These analyses can be performed using the GE computer code SHEX. SHEX is partially based on M3CPT and is used to analyze the period from when the break begins until after peak suppression pool heatup (i.e., the long-term response).

The staff found the assumptions and methods presented in Appendix G of ELTR1 acceptable with the following exceptions:

- The SHEX and TRACG computer codes have not been approved for generic use. However, since the review and acceptance of ELTR1, substantial confirmation of SHEX has been performed by both GENE and the NRC. In accord with the stipulations in Letter to G. L. Sozzi (GE) from A. Thadani (NRC), "Use of the SHEX Computer Program and ANSI/ANS 5.1-1979 Decay Heat Source Term for Containment Long-Term Pressure and Temperature Analysis," dated July 13, 1993 (Reference 18), GENE has performed benchmarking calculations for each application of SHEX. The NRC has performed independent confirmatory analyses on extended uprates for both Mark I and

Mark III containment designs and found the results consistent with SHEX results. Therefore, the confirmatory calculations with SHEX (benchmarking with current licensing basis assumptions – pre-uprate) for plant specific modeling are not required for extended power uprates for Mark I and Mark III containment designs. If future NRC performed independent confirmatory analyses on an extended power uprate for a Mark II containment design also is determined to be consistent with the SHEX results, then similar confirmatory calculations for plant-specific modeling are not required for extended uprates for Mark II containment designs. Plant-specific submittals must continue to show a comparison of SHEX results at the pre-uprate and uprated conditions.

- Plant-specific evaluations should continue to use the computer codes described in the safety analysis report for the facility unless adequate justification for use of an alternate code is provided.

- Safety-relief valve dynamic loads should be evaluated including the effects of second actuation conditions.

In addition to the above analyses, the CPPU LTR describes that the effect of increased decay heat (i.e., increased containment pressure and temperature) on the following equipment will be evaluated [

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The staff reviewed the scope of the proposed containment evaluation for CPPU and found it acceptable. The proposed methodology was previously reviewed and accepted for plant-specific applications provided that suitable justification for its use and validation of its accuracy are included with each plant-specific analysis.

4.2 Emergency Core Cooling Systems

The ECCS components are designed to provide protection in the event of a LOCA due to a rupture of the primary system piping. The limitation of constant reactor dome pressure minimizes the effect of power uprate for ECCS evaluation. Although design basis accidents (DBAs) are not expected to occur during the lifetime of a plant, plants are designed and analyzed to ensure that the radiological dose from a DBA will not exceed the 10 CFR Part 100 limits. For a LOCA, 10 CFR 50.46 specifies design acceptance criteria based on: (a) the peak cladding temperature (PCT), (b) local cladding oxidation, (c) total hydrogen generation, (d) coolable core geometry, and (e) long-term cooling.

The LOCA analysis considers a spectrum of break sizes and locations, including a rapid circumferential rupture of the largest recirculation system pipe. Assuming a single failure of the ECCS, the LOCA analysis identifies the break sizes that most severely challenge the ECCS systems and the primary containment. The MAPLHGR operating limit is based on the most limiting LOCA analysis, and fuel vendors perform LOCA analyses for each new fuel type to demonstrate that the 10 CFR 50.46 acceptance criteria can be met.

The ECCS components considered for CPPU 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 automatic depressurization system (ADS). The following topics are addressed:

- High Pressure Coolant Injection
- High Pressure Core Spray
- Core Spray or Low Pressure Core Spray
- Low Pressure Coolant Injection System
- Automatic Depressurization
- ECCS Net Positive Suction Head

4.2.1 <u>High Pressure Coolant Injection</u>

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 to the SRV setpoints. 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 of the CPPU LTR.

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, as described in Section 9.1 of the CPPU LTR. The adequacy of the HPCI system to meet the safety requirement following a loss of feedwater flow event is discussed in Section 9.1.3 of the CPPU LTR.

[

].

The staff agrees that this [

], the plant must perform a [ ] evaluation using an approved methodology. This approach is acceptable to the staff.

4.2.2 <u>High Pressure Core Spray</u>

The HPCS system (with other ECCS systems as backup) is designed to maintain reactor water level inventory during small and intermediate-break LOCAs, isolation transients and LOFW. The HPCS system is designed to pump water into the reactor vessel over a wide range of reactor operating pressures. The HPCS system also serves as a backup to the RCIC system. The system is designed to operate from normal offsite auxiliary power or from its dedicated emergency diesel generator.

The HPCS system is required to start and operate reliably over its design operating range. During the LOFW event and isolation transients, the RCIC maintains water level above the top-of-active-fuel level (TAF). For the MSIV closure, the SRVs open and close as required to control pressure and the HPCS eventually restores water level.

The capability of the HPCS system during operation at the CPPU power level must be evaluated to ensure core cooling to the reactor in order to prevent excessive fuel PCT following small-and intermediate-break LOCAs, and to ensure the system capability to restore core coverage up to the TAF in isolation transients and LOFW transients. The HPCS evaluation should be applicable to and consistent with the evaluation in Section 4.3 of ELTR2. The maximum reactor pressure at which the HPCS system must be capable of injecting into the vessel for the RCIC backup function is typically selected based on the upper analytical values for the second lowest-set group of SRVs operating in the low-low setpoint mode of operation. [ ].

The generic evaluation in Section 4.3 of the Supplement to ELTR2 is based on typical HPCS pump design pressures. The licensee must evaluate the capability of the HPCS system to perform as designed and analyze its performance at the CPPU conditions, and confirm that HPCS system can start and inject the required amount of coolant into the reactor for the range of reactor pressures associated with LOCAs and isolation transients. CPPU does not change the power required for the pump or the power required from the dedicated HPCS diesel generator.

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 ELTR-2. The adequacy of the system is discussed in the CPPU LTR Section 4.3 and in the containment evaluation, Section 4.1.

The HPCS system serves as a backup to the RCIC system to provide makeup water in the event of a loss of feedwater flow transient. The safety requirement following a loss of feedwater flow event is discussed in Section 9.1.3 of the CPPU LTR.

The SRV settings remain the same for CPPU, [.]

The staff accepts the proposed approach to demonstrate HPCS capability under CPPU conditions because the ECCS-LOCA analysis discussed in Section 4.3 of the CPPU LTR is based on the current HPCS capability, and [ ] evaluations at CPPU conditions will demonstrate that the system provides adequate core cooling.

4.2.3 Core Spray or Low Pressure Core Spray

The LPCS system initiates automatically in the event of a LOCA and in conjunction with other ECCS systems.

As indicated in the ECCS performance discussion in Section 4.3 of the CPPU LTR, the calculated LOCA PCT could increase slightly due to the CPPU. However, the existing LPCS system, combined with other ECCS systems, should still provide adequate long-term post-LOCA core cooling. The existing LPCS system hardware has the capability to perform its design injection function at the CPPU conditions and the generic evaluation in Section 4.1 of ELTR2 should bound the CPPU LPCS system performance. The ECCS-LOCA analysis (see

Section 4.3 of the CPPU LTR) is based on the current LPCS capability, and will confirm on a plant-specific basis that the system provides adequate core cooling. The staff further reviewed ECCS system performance, as discussed in Section 4.3 of this SE. The staff finds the proposed evaluation and confirmation approach acceptable.

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 discussed in Section 4.3 of the CPPU LTR. There is no expected change in the reactor pressure at which the CS/LPCS is required. [ ].

The staff agrees that this [ ], the plant must perform a [ ] evaluation using an approved methodology. This approach is acceptable to the staff.

4.2.4 Low Pressure Coolant Injection

The LPCI mode of the RHR system is automatically initiated in the event of a LOCA and, in conjunction with other ECCS systems, the LPCI mode is used to provide adequate core cooling for all LOCA events. The licensee will confirm that the existing system has the capability to perform the design injection function of the LPCI mode for operation at the CPPU condition and that the generic evaluation in Section 4.1 of ELTR2 [.] Since the ECCS-LOCA analysis (see Section 4.3 of the CPPU LTR), based on the current LPCI capability will demonstrate that the system provides adequate core cooling, the staff finds the proposed approach acceptable.

The LPCI mode of the RHR system is automatically initiated in the event of a LOCA. There is no change in the reactor pressures at which the LPCI mode of RHR is required. 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 adequacy of this system is discussed in Section 4.3 of the CPPU LTR.

The staff agrees that this [ ], the plant must perform a [ ] evaluation using an approved methodology. This approach is acceptable to the staff.

4.2.5 Automatic Depressurization System

The ADS uses relief or safety/relief valves (SRVs) to reduce reactor pressure after a small-break LOCA, allowing the LPCI and CS/LPCS systems to provide cooling flow to the vessel. The adequacy of this system is discussed in Section 4.3 of the CPPU LTR. CPPU does not change the conditions at which the ADS must function. The plant design requires the SRVs to have a minimum flow capacity. After a specified delay, the ADS actuates either on low water level plus high drywell pressure or on sustained low water level alone. The licensee will confirm that the ability of the ADS to initiate on appropriate signals [.] Since the licensee's ECCS-LOCA analysis (see Section 4.3 of the CPPU LTR), based on the current ADS capability, demonstrates that the system provides adequate core cooling, the staff finds the evaluation acceptable.

The staff agrees that this [

], the plant must perform a [ ] evaluation using an approved methodology. This approach is acceptable to the staff.

4.2.6 <u>Emergency Core Cooling System Net Positive Suction Head</u>

Operation at CPPU conditions increases the reactor decay heat, which increases the heat addition to the suppression pool following a LOCA event. As a result, the long-term peak suppression pool water temperature and long-term peak containment pressure may increase. The most limiting case for NPSH typically occurs at the peak long-term suppression pool temperature. The ECCS NPSH was evaluated in Section 4.1.8.5 of ELTR2, Supplement 1, Volume I. 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 of the CPPU LTR). If these values are bounded by the previous evaluation, no additional plant-specific analyses are required for the NPSH.

[

.] The CPPU LTR states that the ECCS NPSH evaluation will be based on the methodology described in Section 4.1.8.5 of ELTR2. This approach involves a <u>plant-specific</u> analysis of the effect of the increased wetwell temperature on NPSH. To the extent credited in the current design basis, the approach credits positive containment pressure to augment NPSH. The staff finds this approach acceptable. However, if, due to the effects of power uprate, this positive containment pressure is credited for a longer duration or a higher magnitude, then these changes would be subject to additional review.

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 under all LOCA conditions and the analysis models must satisfy the requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K.

The CPPU approach takes an exception to the guidelines given in ELTR1. The ELTR1 approach called for a complete plant-specific break spectrum evaluation to be submitted as part of the PUSAR, using equilibrium core design parameters. In the CPPU approach, the LOCA analysis description is based on [

.]

The CPPU approach [ ] is judged to be acceptable for the following reasons:

(a) The staff evaluations of several requests for stretch power increases (i.e., those limited to 5 percent of originally licensed power) and for extended power uprate at BWRs have shown that the change in PCT for power uprates is not significant. The maximum increase in the PCT observed was [ ], and this is well within the acceptance criteria of 10 CFR 50.46. [.]

(b) [.]

(c) The limiting break sizes are well known and have been shown not to be a function of reactor power level.

(d) [.]

(e) The PCT for the limiting large-break LOCA is determined primarily by the hot bundle power, [.]

(f) The reload evaluation confirms that the MAPLHGR for each fuel type in the specific reload core is bounded by the MAPLHGR used in the ECCS-LOCA performance analysis.

(g) If the plant is MAPLHGR-limited or if the LOCA analysis results are at (or above) the acceptance criteria limits, a detailed plant-specific analysis for the licensing basis PCT will be performed.

The LOCA analysis for CPPU builds on the existing SAFER/GESTR LOCA analyses for a plant. [.] The licensing basis PCT is based on the Appendix K PCT. [.] Use of the most limiting of the nominal or Appendix K PCT changes for the licensing basis PCT will ensure continued compliance with the requirements for the SAFER/GESTR LOCA application methodology as approved by the NRC.

In addition to the large-break LOCA analysis, the analysis of small recirculation break LOCA response at CPPU conditions will be reviewed in order to assure adequate ADS capacity. The increased decay heat associated with CPPU will increase the steam generation rate. The higher steam generation rate may result in a longer ADS blowdown and a higher PCT for the small break LOCA. A spectrum of small breaks will be analyzed in order to determine the effect of CPPU on the PCT for the small break LOCA response. A sufficient number of break sizes will be analyzed to establish the worst small break size at CPPU conditions.

The licensee will perform the LOCA analysis using the CPPU methodology at 102 percent of the CPPU rated thermal power (RTP), using the limiting fuel design. The ECCS-LOCA analysis is based on the NRC-approved methodology (i.e., SAFER/GESTR). [

.]

If the CPPU SAFER/GESTR LOCA analysis has sufficient margin to the acceptance criteria, the simplified CPPU analysis approach will be used for the CPPU. [

.] The results will confirm that the licensing basis PCT and upper bound PCT meet the 10 CFR Part 50.46 acceptance criteria and the NRC SE requirements on the SAFER/GESTR LOCA application methodology at CPPU conditions.

The CPPU is expected to [ ] acceptance criteria of 10 CFR 50.46 (local cladding oxidation, core-wide metal-water reaction, coolable geometry). Long-term cooling is assured when the core remains flooded to the jet pump top elevation and when a core spray system is operating.

Because the licensee will perform [ ] evaluations of ECCS-LOCA performance and confirm the applicability [ ] at the CPPU conditions, using approved methods, the staff agrees that the CPPU ECCS-LOCA performance will comply with 10 CFR 50.46 and Appendix K requirements.

The applicability of the [ ] assessment presented in the CPPU LTR will be confirmed in the licensee's submittal. If during the plant specific review it is determined that the [ ] assessment is not applicable to the plant, the plant must perform a [ ] evaluation using an approved methodology. The reload evaluation confirms that the MAPLHGR for each fuel type in the specific reload core is bounded by the MAPLHGR used in the ECCS-LOCA performance analysis. This process is acceptable to the staff.

4.4 Standby Gas Treatment System

In the LTR, GENE identified the CPPU effects on the standby gas treatment system (SGTS). [

.]

The core inventory of iodine and subsequent loading on the SGTS filters or charcoal adsorbers are affected by the power uprate. 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 particulate and halogens, the SGTS limits off-site doses following a postulated design basis accident.

The design flow capacity of the system maintains the secondary containment at the required negative pressure to minimize the potential for exfiltration of air from the reactor building. [

.]

[.] The total post-design basis accident iodine loading on the charcoal adsorbers increases proportionally with the increase in core iodine inventory. Adequate charcoal mass is typically present so that post-accident iodine loading on the charcoal remains within the guidance limits provided in RG 1.52, "Design, Testing, and Maintenance Criteria for Post-Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants."

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 within operating temperature limits is within the cooling flow capability of the system.

The CPPU LTR, as supplemented, presents the assumptions used in and the results of [ ] analyses to evaluate system performance for the following two accident source term models: (1) facilities implementing RG 1.183, "Alternative Radiological Source Term for Evaluating Design Basis Accidents at Nuclear Power Reactors," and (2) facilities that remain committed to RG 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors." The LTR states that the assumptions used in the evaluations [.] The LTR also states that [.]

The analyses demonstrate that the peak iodine loading and charcoal bed temperatures are within the criteria specified in RG 1.52. Therefore, for plants bounded by parameter values used in the [ ] analysis, SGTS performance with regard to iodine loading and charcoal bed temperature is acceptable.

4.5 Conclusion

The CPPU LTR provides an acceptable scope and approach for plant-specific license amendment requests for constant pressure power uprates. Within the scope of the review described in this portion of the staff's evaluation, and when relevant assumptions are satisfied, the staff found that [ ] evaluations for the following systems and design basis events were acceptable: the standby gas treatment system, the liquid and solid waste management systems, the gaseous waste management system, the main steamline flow restrictors, the main steamline break mass and energy release and associated evaluations, and the flooding analysis for moderate energy line breaks. These [ ] evaluations provide either a new design basis for a system or a basis for continued validity of existing design basis evaluations for operation at the uprated power level.

5.0 INSTRUMENTATION AND CONTROL

This section addresses the evaluations in RG 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Chapter 7, that are documented in previous plant power uprate submittals. The principal instrumentation and control evaluations are as follows:

- NSSS Monitoring and Control
- Reactor Protection System (RPS)/Engineered Safety Features Actuation System (ESFAS) Instrumentation Trip Setpoint and Allowable Values

- BOP Monitoring and Control
- Technical Specification Instrument Setpoints

5.1 Nuclear Steam Supply System and Balance-of-Plant Monitoring and Control Systems

The instruments that monitor and the controls that directly interact with or control reactor parameters are usually within the NSSS. TSs address those instrument allowable values and/or setpoints for those NSSS sensed variables, which initiate protective actions. The effect of CPPU on TSs is addressed in Section 5.3 of the CPPU LTR. The topics considered in this section are:

- Average Power Range Monitors, Intermediate Range Monitors, and Source Range Monitors
- Local Power Range Monitors
- Rod Block Monitor
- Rod Worth Minimizer/Rod Control Information System

For CPPU, GENE evaluated the NSSS and BOP systems [.] The plant-specific submittal will [ ] provide a [ ] evaluation. The staff agrees with the GENE evaluation. However, the staff believes that the evaluation should go beyond the systems analysis and should cover all CPPU related changes to instrumentation and controls (setpoint and scaling changes, changes to upgrade obsolescent instruments, changes to the control philosophy).

In response to the staff's concern, GENE submitted a response by letter dated December 3, 2001 (Reference 3), stating that any plant system design that falls outside the basis for the generic analysis will be addressed in the plant-specific submittal. GENE has committed to include these requirements in the CPPU PUSAR shell which is used as the starting point in the preparation of plant-specific CPPU PUSAR documents and reflects the expected level of details for each section. Any major changes to the NSSS or BOP monitoring and control are addressed in the plant-specific CPPU PUSAR. Based on this commitment, the staff finds the proposed approach acceptable.

5.2 Reactor Protection System/Engineered Safety Features Actuation System Instrumentation Trip Setpoint and Allowable Values

In ELTR2, GENE committed to use their instrumentation setpoint methodology in Reference 14 to determine instrument setpoints for the RPS and ESFAS instrumentation. The staff has previously reviewed this instrument setpoint methodology and found it acceptable for establishing new setpoints in power uprate applications. However, GENE has proposed to use a simpler method for determining instrument setpoints in the CPPU LTR. GENE justified this change based on the experience with other extended power uprate applications [

.] In its request for additional information, the staff expressed the concern that the proposed method may not be consistent with the plant's licensing basis and may result in non-conservative instrument setpoint and allowable values. In its response, GENE provided the following restrictions on the use of the simplified process to assure its validity:

- NRC-approved GENE or plant-specific setpoint methodology is used.

[ ]

[

.]

In addition to imposing these restrictions, GENE justified the application of this methodology to the following seven instrument setpoints:

[

]

GENE has justified each of these instrument setpoints and allowable values based on the fact that either the CPPU has no effect on instrumentation error or is not credited in the accident analysis, or the magnitude of the error has no effect on the analysis. GENE provided enough basis to demonstrate for each of these instrument setpoints that the simplified method will not have any effect on the plant's licensing basis [

.] Therefore, the staff finds the simplified instrument setpoint methodology discussed in the LTR for extended power uprate acceptable.

5.3 BOP Monitoring and Control

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 of the CPPU LTR and is considered in Section 10.1.1 of the staff's SE.

5.4 Technical Specification Instrument Setpoints

TS instrument allowable values and/or setpoints are those sensed variables that initiate protective actions and are generally associated with the safety analysis. The determination of instrument allowable values and setpoints generally includes consideration of measurement uncertainties and is derived from the conservative analytical limits used in specific licensing or safety evaluations. Increases resulting from CPPU in the core thermal power and steam flow affect some instrument setpoints. The following setpoints are discussed in this section:

- APRM Flow-Biased Scram
- Rod Worth Minimizer/RCIS Rod Pattern Controller Low Power Setpoint
- Rod Block Monitor
- RCIS Rod Withdrawal Limiter High Power Setpoint
- APRM Setdown in Startup Mode

5.4.1 <u>APRM Flow-Biased Scram</u>

[

.] The TS will be modified by adjusting the flow-biased scram setpoint.

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

he 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. [

.]

5.4.3 Rod Block Monitor

The severity of rod withdrawal error during power operation event is dependent upon the RBM rod block setpoint. [

.]

5.4.4 RCIS Rod Withdrawal Limiter (RWL) High Power Setpoint (HPSP)

[

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5.4.5 APRM Setdown in Startup Mode

The value for the TS 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 of the CPPU LTR. The setpoint for the APRM setdown in the startup mode is based on the TS setpoint. The current TS may be based on either a conservative generic setpoint or on a plant-specific calculated value.

5.5 Conclusion

GENE has justified each of these instrument setpoints and allowable values based on the fact that either the CPPU has no effect on instrumentation error or is not credited in the accident analysis or the amount of error has no effect on the analysis. The staff reviewed the CPPU LTR discussion and finds the simplified setpoint methodology for these instruments under the conditions specified in the LTR acceptable. This is based on the staff's expectation that licensees referencing the CPPU LTR will justify any plant-specific differences from the CPPU LTR with respect to instrumentation setpoint methodologies.

Based on the above review and evaluation of the LTR and GENE's responses to the staff's RAI, the staff concludes that instrument setpoint changes for CPPU are acceptable.

6.0 ELECTRICAL POWER AND AUXILIARY SYSTEMS

The power system includes the generators, the main transformers, the switchyard, and the other transformers. Each generator is connected through a forced cooled isolated phase bus to main transformers. Alternating current (AC) power to the onsite distribution system is provided from the main generator output, from the transmission lines, or from onsite diesel generators.

6.1 AC Power

6.1.1 Grid Stability

The increased power from the generator may affect grid stability/reliability. For CPPU, GENE proposed to handle this effect by administrative controls or distribution logic. The staff raised concerns that the increase in megawatt (MW) electrical due to the power uprate would affect the supply of reactive power (MVAR) from the main generator. The staff considered that a decrease in MVAR output from the main generator affects the voltage on the grid and the voltage to the Class 1E systems and the MVARs cannot be handled by administrative controls or distribution logic alone, as stated in the CPPU LTR. In response to the staff's request for additional information on the grid stability analysis, GENE stated that the CPPU PUSAR shell identifies the requirement that a grid stability analysis be performed, and the results of the analysis are to be summarized in the plant-specific submittal. In addition, GENE has added the following sentence to the CPPU Basis of Section 6.1:

> "The licensee will perform a grid stability analysis, and the results of the analysis will be summarized in the plant-specific submittal. Any plant changes to control the reactive power will be identified in the plant-specific submittal."

Also, GENE has deleted the following sentence from the CPPU Basis Section 6.1:

> "However, this effect can be handled by administrative controls or distribution logic."

The staff concluded that the above response from GENE satisfies the staff's concern. There is reasonable assurance that GDC-17 will be met at the uprated power condition, assuming the plant-specific submittal appropriately addresses the areas described above.

6.1.2 Main Generator

The turbine generator converts the thermal energy in the steam into electrical energy. 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. Experience with previous power uprate applications indicates that turbine generator modifications may be required to support power uprate. The staff raised a concern that the increase in generator MW would also affect the protective relaying for the main generator/main transformer.

In response to the staff's concern on the increase in generator MW and the impact on the protective relaying for the main generator, GENE has added the following sentence to the CPPU basis of Section 6.1:

> "The protective relaying for the main generator may require changing. Any changes will be identified in the plant-specific submittal."

The staff concluded that the above response from GENE satisfies the staff's concern. The staff's review determined that the main generator can be operated safely at the CPPU condition with any need for protective relay changes evaluated in the plant-specific submittal.

6.1.3 Main Power Transformer

The normal loads are increased to support the increased feedwater and cooling water requirements. Experience with previous power uprate applications indicates that main transformer modifications may be required to support power uprate. The staff raised a concern that the increase in generator MW and non-Class 1E loads will also affect the protective relaying for the main transformer and will require minor modifications to ensure reliable operation before achieving full power uprate. In response to the staff's concerns, GENE stated that any main transformer relay changes will be identified in the plant-specific submittal. The staff's review determined that, by modifications to the main power transformer and protective relaying scheme, the main power transformer can be operated safely at the CPPU condition, assuming the plant-specific submittal appropriately addresses this issue.

6.1.4 Isolated Phase Duct

The isolated phase bus conductors and insulators are protected and shielded by continuous, welded aluminum enclosures. Experience with previous power uprate applications indicates that isolated phase duct modifications such as isolated phase bus cooling may be required to support the additional loads associated with CPPU. The staff's review determined that plant-specific submittals will need to evaluate the need for isophase bus cooling modifications.

6.1.5 AC Distribution

The normal loads are increased to support the increased feedwater and cooling water requirements. The staff raised a concern that the type of increased loads such as recirculation pumps, condensate pumps, condensate booster pumps etc. should be addressed. In response to the staff's concern, GENE added the following sentence to the CPPU basis of Section 6.1:

> "The increased normal operating loads are dependent on the specific plant design and may include: the recirculation pumps, condensate pumps, condensate booster pumps, motor drive feedwater pumps, and circulating water pumps."

The staff concluded that the above response from GENE satisfies the staff's concern. Plant-specific submittals will evaluate the increased normal operating loads.

6.1.6 Emergency Diesel Generators

Station loads under emergency conditions are based on existing equipment operating at or below the nameplate rating and within the calculated brake horse power (BHP) for the stated pumps. [

.] Therefore, a plant-specific evaluation of the AC power system is to be performed for CPPU to assure an adequate AC power supply to safety-related systems. The staff finds this to be acceptable.

6.2 Direct Current (DC) Power

GENE stated that experience with previous power uprates has shown that the DC loads are not significantly increased because of power uprate. System loads are computed based on equipment nameplate data. [ ], the DC power distribution system is adequate for CPPU.

The staff concluded that [ ], the design for DC power systems is acceptable for CPPU.

6.3 Standby Liquid Control System

The standby liquid control system (SLC) is a manually operated system that pumps concentrated sodium pentaborate solution into the reactor vessel in order to provide neutron absorption. It is designed to be capable of bringing the reactor to a subcritical shutdown condition from rated thermal power.

An increase in the core thermal power does not by itself directly affect the ability of the SLC boron solution to bring the reactor subcritical and to maintain the reactor in a safe-shutdown condition. A higher fuel batch fraction, a change in fuel enrichment, or a new fuel design may affect the shutdown concentration, but operating at the CPPU condition does not affect the required boron solution. The SLC system shutdown capability is reevaluated [ ]. The effect of the CPPU on the SLC system injection and shutdown capability will be evaluated [.]

The SLC system is designed to inject at a maximum reactor pressure equal to the upper analytical setpoints for the lowest group of SRVs operating in the relief mode. Since the reactor dome pressure [ ] will not change, [.] The SLC pumps are positive displacement pumps, and small changes in the SRV setpoint would have no effect on the SLC system capability to inject the required flow rate. The licensee will confirm [ ] that there is sufficient margin to lifting the SLC system relief valves. The calculated maximum required pump discharge pressure, based on the peak reactor pressure during the limiting ATWS event, should be below the lowest calculated nominal opening pressure for the SLC pump relief valves. Consequently, the SLC relief valves would not lift during the ATWS events. The operation of the SLC system is also analyzed to confirm that the pump discharge relief valves will reclose in the event that the system is initiated before the time that the reactor pressure recovers from the first transient peak. The evaluation compares the calculated maximum reactor pressure needed for the pump discharge relief valves to reclose with the lower reactor pressure expected during the time the SRVs are cycling opened and closed. Considerations are also given to system flow, head losses for full injection, and cyclic pressure pulsations due to the positive displacement pump operation in determining the setpoint for the relief valve. The relief valves are periodically tested to maintain this tolerance. Otherwise, it is expected that the relief valves will operate as designed and originally tested.

The SLC ATWS performance is addressed in Section 9.3.1 of the LTR and the licensee will confirm that the evaluation was based a representative core design at the CPPU condition. The minimum allowable solution concentration used in the ATWS analysis may be increased from the current value. This may be done to minimize the risk of having the ATWS analysis for CPPU generate a peak suppression pool temperature that exceeds current design limits.

6.4 Conclusion

The staff reviewed the information provided in the CPPU LTR, along with responses to the staff's concerns that described the plant-specific evaluations expected for CPPU and the generic effects on parts of the electrical power system. The staff concludes that CPPU would not adversely affect the plant electrical power systems and is, therefore, acceptable.

Based on the description and evaluation of the system operation that will be provided by licensees pursuing CPPU, the staff agrees with the conclusion that the SLC will be able to inject boron into the reactor coolant system as required by 10 CFR 50.62.

7.0 POWER CONVERSION SYSTEMS

7.1 Turbine Generator

The CPPU LTR identifies a turbine missile evaluation for plants using turbines with shrunk-on wheels as the only necessary safety-related evaluation associated with turbine operation at the uprated power level. As turbines with integral rotors are not considered a credible source of turbine missiles, it is not necessary to evaluate the potential change in turbine missile frequency at plants using this type of turbine. In their supplement provided by letter dated December 3, 2001, GENE stated that although the power uprate slightly increases energy trapped in the turbine following a load rejection, the turbine overspeed would remain within design limits. Therefore, this limitation in the scope of turbine generator reviews for power uprates is acceptable to the staff.

7.2 Turbine Steam Bypass System

The turbine steam bypass system is a normal operating system that is used to bypass excessive steam flow. The bypass flow capacity is included in some AOO evaluations (Section 9.1 of the CPPU LTR). These evaluations demonstrate the adequacy of the bypass system. The turbine steam bypass system capacity in terms of mass flow is not changed for CPPU. As a result, the increase in reactor power level and the resulting increase in steam flow to the turbine effectively reduces the bypass system capacity in terms of percent of uprated steam flow. 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. This approach is acceptable to the staff.

7.3 Feedwater Control System

The increase in reactor power and steam flow for CPPU results in an increase in feedwater flow. The feedwater control system is a normal operation system that controls the water supply to the reactor to maintain water level. Failure of this system is evaluated in the reload analysis for each reload core with the feedwater controller failure-maximum demand event. This approach is acceptable to the staff. A loss of feedwater event can be caused by downscale failure of the controls. The loss of feedwater flow and loss of one feedwater pump events are discussed in Section 9.1.3 of the CPPU LTR.

8.0 RADWASTE SYSTEMS AND RADIATION SOURCES

This evaluation focuses on the power uprate's impact on occupational worker doses from increases in plant dose rates, and impacts on calculated doses to members of the public from increases in radioactive effluents.

8.1 Liquid and Solid Waste Management

GENE concludes that increased power levels and steam flow result in slightly higher levels of both liquid and solid radwastes. GENE notes that the major impact of the power uprate on liquid and solid radioactive waste production is the increased generation of spent condensate cleanup resins (SCCR). Because of the estimated increased levels of feedwater flow and corrosion products in the feedwater system, SCCR quantities are expected to increase as a result of the increased change-out frequency for resin bed media. Similarly, due to slightly higher levels of activation and fission products in the reactor coolant, the reactor water cleanup (RWCU) filter-demineralizer will require more frequent backwash/change-out.

The LTR describes the condensate demineralizers as the single largest source of liquid and wet solid waste. [ ]. Therefore, the overall increase in liquid and solid waste volume will be bounded by the increase in condensate demineralizer flow rate. The staff concludes that this small increase is within the capability of the system, and, due to the batch nature of the discharges, can be easily managed within regulatory requirements to maintain discharges to the environment as low as reasonably achievable.

[ ], GENE expects the quantity (activity) of activated corrosion products (ACP) to increase [.] The average offsite doses to the public from the liquid release pathway are generally very small fractions of the 10 CFR Part 50, Appendix I, numerical design objectives and the dose limits of 40 CFR Part 190. The projected ranges of increases of these very small calculated doses are expected to result in a negligible increase in calculated public dose, and the overall contribution to the public dose from the liquid effluent pathway would remain a very small fraction of the regulatory limits.

However, due to site-specific environmental factors and plant-specific radwaste equipment configurations and waste management practices, a plant-specific evaluation must be submitted with each power uprate request. These evaluations will assess the operational impact of increased liquid and solid waste processing. The evaluation will ensure that plant liquid effluent releases remain as-low-as-is reasonably achievable (ALARA), and the resultant calculated doses remain below the dose limits of 10 CFR Part 20 and meet the requirements of 10 CFR Part 50, Appendix I.

8.2 Gaseous Waste Management

Gaseous wastes generated during normal operation are collected, controlled, processed, stored, and disposed of using the gaseous waste management (offgas) system. This system is designed to process and control the release of gaseous radiological effluents to the environment such that the total radiation exposure of persons in offsite areas is as low as reasonably achievable. The release rate is administratively controlled to remain within limits and is principally a function of fuel cladding performance, main condenser air inleakage, and charcoal adsorber performance. These factors are not a function of reactor power. However, the power uprate has a secondary effect in that any fuel pin leaks will release greater quantities of fission product gasses and a greater fraction of condenser inleakage will be activated by the higher average neutron flux. But these secondary effects are negligible in comparison with variations in the primary contributors to gaseous radiological effluents.

Radiolysis of water (i.e., formation of H_2 and O_2) in the core increases linearly with power, thus increasing the heat load on the offgas recombiner and related components. Because the offgas recombiner and associated condenser remove most of the radiolysis products from the waste gas stream as liquid water, this increase has a negligible effect on other portions of the offgas system.

[

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Thus, offgas system components designed for the generic rate of radiolytic gas production have ample margin to accommodate the increase in radiolytic gas production associated with the power uprate.

Based on the above, the staff concludes that administrative controls are adequate to manage any increase in gaseous radiological effluents resulting from the power uprate. Also, plants having recombiners and associated components originally designed for a radiolytic gas production rate of [ ] have adequate margin to accommodate the increased radiolysis resulting from the power uprate. However, systems designed to a radiolytic gas production rate lower than [ ] remain subject to plant-specific review, as specified in the LTR.

GENE discussed the main offgas system (MOS) and the impact of the power uprate on the MOS gaseous radiological effluent release rate. Gaseous fission products such as Krypton-85 and Iodine-131 are produced by the fuel in the core during normal reactor operation. A small percentage of these fission gases is released to the reactor coolant from a small number of fuel assemblies which are assumed to develop leaks during reactor operation. GENE estimates that fission product gases input to the MOS will increase [
.] The main offgas system removes these fission gases directly from the plant main condenser, and they are processed before release. The MOS effluent release rate is a function of fuel cladding integrity, air in-leakage into the main condenser, charcoal inlet dew point and absorber temperature. GENE notes that the functions of the MOS and the main offgas radiological gaseous effluent release rate [
.] Given that installed MOS in operating plants effectively reduce gaseous effluents by factors greater than 100, fission product input increases of up to 20 percent into the MOS are expected to have a negligible impact on calculated doses to the public from gaseous effluents.

8.3 Radiation Sources in the Reactor Core

For both plant operation and post-operation, GENE has examined the impact of the power uprate on the facility radiation levels from radiation sources in the core. The radiation sources in the core include radiation from the fission process, accumulated fission products, and neutron reactions as a secondary result of reactor power. The radiation sources in the core during operation are expected to increase in proportion to the increase in power. However, this increase is bounded by the existing safety margins of the design basis sources. Since the reactor vessel (inside the fully-inerted primary containment) is inaccessible during operation, a proportional increase in the radiation sources in the reactor core will have no effect on occupational worker personnel doses during power operations. Due to design shielding and containment surrounding the reactor vessel, worker occupational doses are largely unaffected, and doses to the public from radiation shine from the reactor vessel remain essentially zero as a result of the power uprates. Potential impacts of increased dose rates inside primary containment on component reliability are discussed in Section 10.3 of the staff's safety evaluation.

From a post-operation perspective, GENE discussed the two separate sets of radiation source data for the core, and both must be corrected for radioactive decay after shutdown. The first, the gamma-ray source, is used for radiation shielding calculations for the core and individual fuel bundles. In terms of MeV/sec per reactor thermal power, this source is a function of, and increases in proportion with, reactor power. The second set of post-operational source data is the nuclide activity (fission products primarily) in the fuel. This data is used as input for post-accident and spent fuel analyses, which apply appropriate regulatory modeling for source term release fraction, timing and transport assumptions and parameters. Both short-lived and long-lived nuclides are expected to increase in approximate proportion to increase in core thermal power. GENE discusses appropriate decay and equilibrium considerations, and establishes bounding parameters to be used for core radiation source calculations. [.] Plant power uprate applications that conform with the values of these bounding parameters would be acceptable.

However, as discussed in Section 8.5 of this SE, in order to follow NUREG-0737, Item II.B.2, post-accident shielding requirements, licensees would need to perform plant-specific analyses of post-accident dose rates as they affect operator access to designated vital areas.

8.4 Radiation Sources in the Reactor Coolant

Radiation sources in the reactor coolant contribute to the plant radiation levels. These sources include coolant activation products, activated corrosion products (ACP) and fission products. GENE examined the impact of the power uprate on each type of source. The staff accepts the approach described below to address CPPU effects on radiation sources in reactor coolant.

8.4.1 <u>Coolant Activation Products</u>

During operations, the reactor coolant passing through the reactor core region becomes radioactive as a result of nuclear reactions. GENE notes that the activation product concentrations in the steam [ ] following the power uprate since the increase in activation production in the steam passing through the core is [ ] with the power increase, but [ ] by the increase in steam flow through the core. [ ], the transit time from the core to the turbine building components will be reduced (due to increased steam flow rate). This decrease in transit time reduces the decay period of very short-lived radionuclides (mainly N–16), resulting in higher dose rates, roughly proportional to the power increase, in and around the turbine/condenser and other main steam components.

Because of plant-specific design and varying operational chemistry regimes, the percent increase in activation products (and operational doses rates) as a result of the power uprate will be determined [.]

8.4.2 Activated Corrosion Products

ACPs result from the activation of metallic corrosion and wear materials in the reactor coolant, and are expected to increase as a result of a power uprate. The equilibrium level of ACPs in the reactor coolant is expected to increase as a result of the increase in feedwater flow rate and the increase in neutron flux in the reactor. The increased feedwater flow will likely reduce the efficiency of the condensate filtration and demineralization system (CFDS), thereby resulting in an additional increase in the equilibrium level of ACPs (and increased external dose rates). However, GENE expects that the ACP increase will not exceed the design basis concentrations.

Because of plant-specific design of the CFDS and feedwater systems, and varying operational chemistry regimes, the increase in ACP as a result of the power uprate will be determined by a plant-specific analysis.

8.4.3 Fission Products

Fission products in the reactor coolant result from the escape of minute fractions of the fission products in the fuel rods. Fission product release into the primary coolant is dependent on the nature and number of fuel defects and GENE does not expect an increase in these defects as a result of the power increase. [

.] Given that current levels of fission product activity typically found in reactor coolant and steam are [ ], a percent fission product increase of no more than the power uprate [.]

Because of potential plant design and operational differences, [

.]

8.5 Radiation Levels

External radiation levels contribute to the plant worker occupational doses during plant operation, post operations (plant shutdowns), and during postulated accident conditions. These plant radiation levels result from activation and fission products, and ACP discussed in Section 8.4. GENE examined the impact of power uprates for each operational mode or condition.

GENE stated that many aspects/areas of the plant were conservatively designed for higher-than-expected radiation sources. Therefore, for most plants, the increases in radiation levels during operations at higher power levels will not affect radiation zoning or shielding adequacy for most plant areas. [

], plants that employ hydrogen water chemistry (HWC) experience increasing radiation levels [ ] from gaseous activation products (chiefly N-16) in and around the turbine building during plant operations. The NMIP is used primarily to maintain worker doses ALARA. The NMIP also provides a public dose reduction benefit by significantly reducing hydrogen injection rates which results in the reduction of direct radiation shine from the steam-side turbine building components. Thus HWC-only plants may have little design margin remaining and could be particularly impacted by power uprates. In any case, all plants should perform special surveys and monitor for external radiation level changes during power ascension to ensure any significant radiation level increases in specific areas are identified in a timely manner and controlled in accordance with 10 CFR Part 20 and plant technical specifications. For plants employing the HWC injection process, a plant-specific analysis will be performed to assess the impact of increasing radiation levels from the increased activity level of gaseous activation products on plant radiation zoning in and around the turbine building and main steam piping.

The post-operation external radiation levels [.] ACP are the main source of shutdown dose rates and the chief contributor to occupational doses to workers. Some post-operational radiation levels may also be higher in those areas of the plant where accumulation of ACP is expected (i.e., near the spent fuel pool cooling system piping and the reactor coolant piping as well as near some liquid radwaste equipment). Licensees will use pre-job worker training/briefings, procedural access, and work planning and controls to compensate for any increased radiation levels and to maintain occupational doses ALARA. [ ], applicants also need to analyze the impact of the resultant increase in radiation levels in plant systems and areas from ACP.

GENE concludes that post-accident radiation levels will increase due to the change (increase) in the reactor core source term inventory as a result of power uprate. [.]

This increase in post-accident dose rates impacts Item II.B.2 of NUREG-0737, which establishes occupational worker dose guidelines during a postulated accident so that operators can access and perform required duties and actions in designated vital areas. These design limits (GDC 19, 10 CFR Part 50, Appendix A) require that adequate radiation protection be provided such that the dose to personnel not exceed 5 rem whole body or the equivalent to any part of the body for the duration of the accident (the extremity limit is 75 rem). Plant design and vital area missions differ from plant to plant, and since plants can choose between two different post-accident dose models, a plant-specific evaluation of the power uprate impact on all vital areas and missions will need to be performed to ensure that personnel access to and work in designated vital areas for accident mitigation following a LOCA can still be accomplished without exceeding the dose requirements of GDC 19.

8.6 Normal Operation Off-Site Doses

During normal operation GENE notes that the two primary sources of off-site public doses are airborne releases from the MOS, and gamma shine from the plant steam turbines and associated steam components.

The MOS effluent release quantities are greater than the sum of all gaseous streams released by the plant. GENE estimates that the MOS effluent gaseous releases will increase by no more than the percentage increase in reactor thermal power. Given that the installed MOS in operating plants effectively reduces main offgas effluents by factors greater than 100, effluent release increases of up to 20 percent from the MOS are expected to have a negligible impact on calculated doses to the public. GENE concludes that the actual estimated increase in off-site doses from the MOS will be determined by [ ] to ensure that the public doses remain below the limits of 10 CFR Part 20, 10 CFR Part 50, Appendix I and 40 CFR Part 190.

Gamma radiation (skyshine) from coolant activation products (chiefly Nitrogen-16) in the reactor steam in the main steam system components in the turbine building provides another offsite public dose pathway. GENE notes that the power uprate results in increased steam flow, leading to generally proportional higher levels of activation products (chiefly Nitrogen-16) and resultant external dose rates in and around the turbine building. Typical shielding design more than adequately bounds any such radiation level increase due to power uprate. During power operations, N-16 production is increased by the HWC process (routine hydrogen gas injection into the reactor feedwater in an effort to prevent intergranular stress corrosion cracking of reactor internals). The resulting higher dose rates then increase the gamma skyshine both on-and off-site. Applicants should be aware of the impact on station workers working in buildings adjacent to the turbine building (e.g., administrative station employees that may be designated as members of the public). These station employees then would be subject to the 10 CFR Part 20 public dose limits. For plants that use HWC, a site-specific analysis will need to be performed to confirm that the turbine building skyshine increases due to power uprate do not result in doses to members of the public exceeding the limits in 40 CFR Part 190.

9.0 REACTOR SAFETY PERFORMANCE EVALUATIONS

This section addresses the evaluations in Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Chapter 15, that are documented in the current plant power uprate submittals. These reactor safety performance evaluations include:

- Anticipated Operational Occurrences
- Design Basis Accidents
- ATWS
- Station Blackout

Plant-specific evaluations will be included in the plant-specific submittal consistent with the format and level of detail of previous extended power uprate submittals or as discussed in the CPPU LTR sections. 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.

The staff agrees that this [

], the plant must perform a [ ] evaluation using an approved methodology. This approach is acceptable to the staff.

9.1 Anticipated Operational Occurrences

AOOs are abnormal transients that are expected to occur one or more times in the life of a plant and are initiated by a malfunction, a single failure of equipment, or a personnel error. The applicable acceptance criteria for the AOOs are based on 10 CFR Part 50, Appendix A, GDC 10, 15, and 20.

GDC 10 requires that the reactor core and associated control and instrumentation systems be designed with sufficient margin to ensure that the SAFDL are not exceeded during normal operation and during AOOs.

GDC 15 stipulates that sufficient margin be included to ensure that the design conditions of the reactor coolant pressure boundary are not exceeded during normal operating conditions and AOOs.

GDC 20 specifies that a protection system be provided that automatically initiates appropriate systems to ensure that the specified fuel design limits are not exceeded during normal operating conditions and AOOs.

The SRP provides further guidelines that:

- pressure in the reactor coolant and main steam system should be maintained below 110 percent of the design values according to the ASME Code, Section III, Article NB-7000, "Overpressure Protection;"
- fuel cladding integrity should be maintained by ensuring that the reactor core is designed to operate with appropriate margin to specified limits during normal operating conditions and AOOs;
- an incident of moderate frequency should not generate a more serious plant condition unless other faults occur independently; and
- an incident of moderate frequency, in combination with any single active component failure or single operator error, should not result in the loss of function of any fission product barrier other than the fuel cladding.

A limited number of fuel cladding perforations are acceptable under these guidelines.

The plant UFSAR typically evaluates a wide range of potential transients. Chapter 15 of the UFSAR contains the design basis analyses that evaluate the effects of an AOO resulting from changes in system parameters such as: (1) a decrease in core coolant temperature, (2) an increase in reactor pressure, (3) a decrease in reactor core coolant flow rate, (4) reactivity and power distribution anomalies, (5) an increase in reactor coolant inventory, and (6) a decrease in reactor coolant inventory.

Plant response to the most limiting transients are analyzed each reload cycle and are used to establish the thermal limits. A potentially limiting event is an event or an accident that has the potential to affect the core operating and safety limits.

The generic guidelines for the EPU evaluation (Appendix E of ELTR1) identified the set of limiting transients to be considered in each event category.

The CPPU approach takes an exception to the guidelines given in ELTR1. The staff SE for ELTR1 states that: "- - -the staff agrees with the minimum set of limiting transients to be analyzed, which is contained in Appendix E of ELTR1." [

.]

Plant-specific core reload analyses (i.e., those results documented in the SRLR and COLR) are not submitted with the licensee's power uprate application and are not normally submitted for NRC staff review and approval. The reload analyses are conducted using methods previously reviewed and accepted by the staff. Further, the methods approved for reload analyses specify the acceptance criteria for the transients to be analyzed. The reload evaluation process is documented in GESTAR-II. Existing regulations require licensees to obtain staff approval for changes to analysis methods and acceptance criteria used for reload analyses. In addition, based on previous experience with reviewing EPU analyses for EPUs that maintained a constant reactor dome pressure, the staff does not expect significant differences in the results of such analyses for pre- and post-CPPU conditions. Therefore, the staff determined that further review of the reload analysis methods or results was not necessary for CPPU applications. The staff may choose to audit certain future reload analyses for CPPU applicants.

The following transients in Appendix E of ELTR1 will be evaluated or re-analyzed [

]:

[

.] This is acceptable to the staff.

All of the transients listed in Table E-1 of ELTR1 were considered. The limiting overpressure transient will be analyzed as defined by GESTAR-II and by the ELTR1. [

.] Analyses of the events listed in Table E-1 of ELTR1, for plants pursuing extended power uprates, have confirmed the applicability of the GESTAR-II list of limiting events. [

.]

The limiting events are defined in GESTAR-II and the core reload analysis will be based on approved GESTAR-II methodology. The other events listed in Table E-1 of ELTR1 do not establish the OLMCPR, based on experience and the characteristics of these events, and therefore are not analyzed to establish this limit.

As discussed above, most of the transients listed in Appendix E of ELTR1 will be analyzed [

] these evaluations are not expected to be included in the power uprate license amendment submittal. [

.] The results of the limiting thermal margin event analyses are dependent upon [

.] Based on the experience with previous analyses of power uprate in this area and that the conclusions in the CPPU LTR will be confirmed on a plant-specific basis, this part of the proposed CPPU approach is acceptable to the staff.

[

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- Fuel Thermal Margin Events
- Power and Flow Dependent Limits
- Loss of Water Level Events (Loss of feedwater flow/Loss of one feedwater pump)

9.1.1 <u>Fuel Thermal Margin Events</u>

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[

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9.1.2 <u>Power and Flow Dependent Limits</u>

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9.1.3 Loss of Water Level Events

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For the LOFWF event, transient core cooling is provided by maintaining the water level inside the core shroud above the top of active fuel. Plant-specific analysis will be performed as described in Section 5.3.2 of ELTR1, using the limiting high pressure 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.

Loss of one feedwater pump involves operational considerations, as discussed in Section 3.9 of the CPPU LTR, to avoid 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 need not be re-evaluated on a plant-specific basis for CPPU.

9.2 Design Basis Accidents: Radiological Consequences

This section addresses the radiological consequences of DBAs. The CPPU LTR section entitled, "Design Basis Accidents," addresses the radiological consequences of DBAs. This section, as originally submitted, addressed five DBAs: LOCA, fuel handling accident (FHA), control rod drop accident (CRDA), main steamline break outside containment (MSLBA), and instrument line break (ILBA). [

.] A cross-reference was made to Appendix H of ELTR1 for analysis guidance. [

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9.2.1 Accident Dose Assessment

The staff reviewed the CPPU LTR for aspects related to analyses of the radiological consequences of DBAs. The staff review focused on sections addressing radiation sources in the reactor core (as it relates to DBAs) and design basis accidents. The staff also reviewed other CPPU LTR sections for potential impacts on the radiological consequences of accidents.

The CPPU LTR section entitled, "Radiation Sources in the Reactor Core," addresses the increase in radiation sources in the core due to the CPPU. These source terms are used in both normal operational and post-accident radiological assessments. This part of the staff's evaluation considers only the post-accident applications of this source term. This section identifies [ ] input parameters for a [ ] analysis of the core inventory expressed in units of Ci/MWt. [.]

The staff requested additional information in this area. GENE responded to this request in a letter dated December 3, 2001. Changes to the CPPU LTR were proposed as part of this response. The staff accepts the CPPU approach, including the proposed changes to be incorporated in the final approved version of the CPPU LTR.

9.2.2 Radiation Sources in the Reactor Core

The CPPU LTR addresses the increase in the magnitude of radiation sources in the reactor core due to the CPPU. Included is a description of a core inventory analysis performed by GENE using the industry-accepted ORIGEN code. This analysis was performed using a set of input parameters and assumptions [.] Although the core inventory itself is not tabulated in the CPPU LTR, a table was provided as part of the RAI response and was reviewed by the staff by comparison to other tabulations and found to be acceptable for use in analyzing DBAs.

9.2.3 Radiological Dose Analysis of Design Basis Accidents

The magnitude of the potential radiological consequences of a DBA is proportional to the quantity of fission products released to the environment. The magnitude of the environmental release is a product of the activity released from the core and the transport mechanisms between the core and the effluent release point. In general, the inventory of fission products in the fuel rods, the creation of radioactive materials outside of the fuel by irradiation, and the concentration of radioactive material in the reactor coolant system are directly proportional to the rated thermal power. Thus, an increase in the rated thermal power can be expected to increase the inventory of radioactive material that is available for release. The transport mechanisms are dependent on plant process parameters, such as process stream flows, temperatures, pressures, and may be dependent on the timing of plant protective responses, such as reactor trips, isolation valve closures, and control room isolation. The power uprate and plant modifications associated with the power uprate may affect the assumptions made in previous consequence analyses. [.]

The original language of the CPPU LTR and that in the cross-referenced ELTR1 focused on the increase in core inventory due to the CPPU. However, the staff felt that inadequate attention was paid to the impact of the CPPU on the transport of radioactive materials from the core to the environment. Although the CPPU LTR identified the potential impact of both core inventory and transport changes, the CPPU LTR asserted that: "For most DBAs, the radiological releases under CPPU are expected to increase proportional to the core inventory increase." No further discussion of transport impacts was provided in the CPPU LTR or in ELTR1. The staff believes that this implied disposition of transport impacts may not be adequate and that there are plant-specific impacts that could negate this conclusion. While the staff agrees that maintaining a constant reactor pressure minimizes the impact on transport mechanisms, it does not eliminate the need to consider the full range of plant-specific impacts. In its RAI, the staff provided several plant-specific examples in which the transport could be affected by the power uprate. Many of these examples applied only to facilities that had implemented an alternative source term (AST). Since the CPPU LTR is intended for use by all BWRs, the staff requested the cross-reference to Appendix H be deleted and that a discussion on the need for a plant-specific evaluation of the impact of a CPPU on transport be added to the CPPU LTR. GENE proposed a revision to the CPPU LTR to provide separate sub-sections for facilities using the traditional TID-14844 source term and those using an AST. These sections address the need to consider impacts on transport. The staff finds the proposed revision acceptable.

The staff has reviewed several extended power uprate requests that reference ELTR1. With the exception of one facility that based its analyses on an AST, all analyses involved development of a dose scaling factor [ ]. This dose scaling methodology is consistent with the discussions in ELTR1 and its appendices (or with the discussions in the CPPU LTR), and has been found acceptable in these early uprate requests. Briefly summarized, the analysis approach develops [.]

Appendix H to ELTR1 stated that the basic premise of the power uprate radiological/radiation evaluations is that the existing calculations as shown in the current safety analysis reports may be extended to higher reactor power levels. The staff believes that this premise may not be appropriate for all facilities. The staff's experience in reviewing license amendment requests indicates that some licensees have analyses based on analysis inputs and assumptions that may not be acceptable for analyses at CPPU conditions.

In approving a license amendment request based on the CPPU approach, the staff must make a finding with reasonable assurance that public health and safety will not be impacted by allowing reactor operation at CPPU conditions. Although the validity of extending current analyses, as proposed in ELTR1 Appendix H was accepted by the staff in its approval of ELTR1, the staff has reconsidered its position as it will apply to future uprate requests.

In its response to the staff's concern on this issue, GENE proposed adding a discussion to the template used for generating the plant-specific PUSAR. While the staff finds that the language of the proposed PUSAR template discussion addresses the staff concern, the PUSAR template was not submitted for review and is not covered by this SE. To ensure that this concern will continue to be addressed, the staff will request that applicants for a CPPU amendment confirm that the current plant analyses, to which the scaling factors will be applied, meet the guidance of current versions of applicable regulatory guides and current versions of applicable sections of the SRP, or to propose alternative justification for extending the current analyses to uprated power conditions.

The staff reviewed the [ ] MSLBA and ILBA provided in the CPPU LTR. While the staff generally found the bases[ ] to be acceptable, the staff requested that GENE expand the discussion to identify the significant assumptions [.] This additional language was provided in the RAI response. The staff finds the proposed revisions to be acceptable.

The staff finds that the CPPU LTR provides an acceptable framework for evaluating the impact of a CPPU of up to 20 percent on the potential consequences of DBAs. Given the significant increase in core inventory and the resulting increase in postulated accident radiation doses due to an increase in rated thermal power of up to 20 percent, the staff expects that each licensee for a power uprate that references the CPPU LTR will perform the required plant-specific evaluations in accordance with the content of the CPPU LTR and the following additional guidance.

a. Plant-specific radiological dose analyses should be performed using the guidance of current versions of applicable RGs and the current versions of applicable sections of the SRP except where the licensee can show that the staff has explicitly accepted an alternative approach to the guidance set forth in the SRP and applicable regulatory guides. Applicants may propose alternatives to the applicable guidance for staff consideration as part of CPPU license amendment requests.

The [ ] is an acceptable methodology provided that:

(1) The licensee has confirmed that the CPPU has not affected analysis inputs [,] and

(2) The prior analyses were performed as stated in Paragraph (a) above.

The CPPU LTR provides [ ] of the core inventory and the impact on the main steam line break accident outside of containment and an instrument line break accident. These [ ] are acceptable to the staff provided that applicants referencing these [ ] confirm that the dispositions are bounding for their facilities.

9.3 Special Events

This section considers two special events: ATWS and Station Blackout (SBO).

9.3.1 Anticipated Transients Without Scram

An ATWS is defined as an AOO with failure of the reactor protection system to initiate a reactor scram to terminate the event. The requirements for ATWS are specified in 10 CFR 50.62. This regulation requires BWRs to have the following mitigating features for an ATWS event:

- A standby liquid control (SLC) system with the capability of injecting a borated water solution with reactivity control equivalent to the control obtained by injecting 86 gpm of a 13 weight percent sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a reactor vessel with 251-inch inside diameter.

- An alternate rod insertion (ARI) system that is designed to perform its function in a reliable manner and that is independent from sensor output to the final actuation device.

- Equipment to trip the reactor coolant recirculation pumps automatically under conditions indicative of an ATWS.

BWR performance during an ATWS is also compared to the criteria used in the development of the ATWS safety analyses described in Reference 15. The criteria include:

- limiting the peak vessel bottom pressure to less than the ASME Service Level C limit of 1500 psig,
- ensuring that the peak cladding temperature remains below the 10 CFR 50.46 limit of 2200 degrees F,
- ensuring that the cladding oxidation remains below the limit in 10 CFR 50.46,
- limiting peak suppression pool temperature to less than the containment design temperature, and
- limiting the peak containment pressure to a maximum of 110 percent of the containment design pressure.

The ATWS analyses assume that the SLC system will inject within a specified time to bring the reactor subcritical from the hot full power condition and maintain the reactor subcritical after the reactor has cooled to the cold shutdown condition.

The plant-specific analysis will confirm that the CPPU conditions meet the ATWS mitigation requirements defined in 10 CFR 50.62, because (a) an ARI system is installed, (b) the boron injection capability is equivalent to 86 gpm, and (c) an automatic ATWS-recirculation pump trip (RPT) has been installed. Section L.3 of ELTR1 discusses the ATWS analyses and provides a generic evaluation of the following limiting ATWS events in terms of overpressure and suppression pool cooling: (a) MSIV closure, (b) pressure regulator failure to open (PRFO), (c) loss-of-offsite power (LOOP), and (d) inadvertent opening of a relief valve (IORV). The ATWS analyses will be performed [ ] at the CPPU operating condition [ ]. The LOOP event will be analyzed only if it produces a significant loss in suppression pool cooling. The plant specific ATWS analysis for CPPU will demonstrate that the ATWS acceptance criteria for [.] To benchmark the plant response to limiting ATWS events at CPPU conditions, the licensee may also perform the ATWS analyses for the current rated thermal power.

[.] The plant operating staff will be able to identify and respond to an ATWS event as under the current power level. [.] Boron injection is assumed to start at the boron injection initiation temperature or 2 minutes after the ATWS trip point (i.e, low reactor water level or high reactor pressure), whichever is later. In both the current power conditions and CPPU conditions, the SLC pumps are started at 2 minutes after the trip point. [.]

The licensee is expected to list the key input parameters used in the plant-specific ATWS analyses and the corresponding results [.] The licensee will confirm that the results of the plant-specific ATWS analyses meet the ATWS acceptance criteria. [

.] Therefore, the proposed approach to evaluation of the plant response to an ATWS event for CPPU operation is judged to be acceptable.

Since the ATWS analyses are based on NRC-approved methods and the licensee will perform the ATWS analyses at the CPPU conditions, the staff accepts the CPPU LTR evaluation approach.

The staff agrees that licensees following the CPPU approach can continue to satisfy the requirements of the ATWS rule stipulated in 10 CFR 50.62 and that the results of the ATWS analyses for CPPU operation are likely to meet the ATWS acceptance criteria. [

.]

9.3.2 <u>Station Blackout</u>

The plant response to and the coping capabilities for the SBO event are affected by operation at the CPPU power uprate level, due to the increase in the decay heat. For CPPU, the SBO event will be reevaluated using the guidelines of NUMARC 87-00 and RG 1.155, "Station Blackout," consistent with the plant-specific licensing basis. The results of the plant-specific evaluation will be reported in the plant-specific submittal. The staff concluded that, if the results of the plant-specific evaluations are consistent with the guidelines of NUMARC 87-00 and RG 1.155, the design for SBO would meet GDC 17 and 10 CFR 50.63 and would be acceptable.

10.0 <u>ADDITIONAL ASPECTS OF CPPU</u>

10.1 High Energy Line Breaks (HELB)

10.1.1 <u>Main Steam Line Break</u>

For CPPU, reactor vessel dome pressure is unchanged relative to the pre-uprate value. Therefore, during plant operation the uprate will not significantly affect the fluid conditions, i.e., pressure, temperature, and enthalpy, within the system piping. In the event of a steam line break, flow will be choked either at the break or at the steam line flow restrictions.
For flow choked at the steam line flow restrictor, the reactor will be brought subcritical by the increased void fraction in the core and the main steam line high flow isolation signal will close the MSIVs. For breaks where flow is choked at the break, high area temperature or high area differential temperature isolation signals initiate MSIV closure. Initiation of MSIV closure initiates a reactor trip, which ensures the reactor is subcritical prior to isolation. During the short period between the reactor going subcritical and the automatic closure of the MSIVs, [

.]

Based on the above information, for plants implementing CPPU, [

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10.1.2 Liquid Line Break

Operation at the uprated power level will affect liquid pressure, temperature, and flow rates within certain piping systems, including the reactor recirculation system and the main feedwater system. Therefore, plant-specific evaluations of HELBs in liquid systems are necessary. The evaluations include subcompartment pressure and temperature, pipe whip, jet impingement, and flooding, consistent with the plant licensing basis, as described in the LTR.

10.2 Moderate Energy Line Breaks

Operation at the uprated power level [

]

reevaluation of flooding events within the plant's current licensing basis would only be necessary for modifications that increase the flow rate in open-cycle systems, such as circulating water.

10.3 Environmental Qualification (EQ) of Electrical Equipment

The CPPU SAR will provide a review of the equipment qualification for safety-related electrical equipment for CPPU [ ] for the normal and accident conditions expected in the area where the EQ equipment is located. [

.]

EQ for safety-related electrical equipment located inside the containment is based on MSLB and DBA/LOCA conditions and their resultant temperature, pressure, humidity and radiation consequences, and includes the environments expected to exist during normal plant operation. The current accident conditions for temperature, pressure, and humidity from containment analysis [.] Normal temperatures and radiation levels are expected to increase slightly in some areas (e.g., near the feedwater lines). [ ] changes in normal and accident conditions to assure EQ equipment remains qualified in the uprated power environment and that documentation of the qualification of equipment is maintained. [.] The plant-specific submittal should address this effect and the NRC staff will review the submittal against the requirements of 10 CFR 50.49. EQ for safety-related electrical equipment located outside the containment is based on MSLB in the steam tunnel, or other HELBs, whichever is limiting for each plant area for accident temperature, pressure, and humidity environments. [

.] The plant-specific submittal should address this effect and NRC staff will review the submittal against the requirements of 10 CFR 50.49. The staff reviewed the information provided in the CPPU LTR, along with responses to the staff's concerns. The staff concludes that CPPU would not adversely affect the plant electrical power systems and the environmental qualification of electrical equipment. The design will conform to GDC 17 and 10 CFR 50.49 and the proposed LTR is, therefore, acceptable.

10.4 Risk Implications

The staff reviewed the individual plant evaluation (IPE) section, which is herein also referred to as the probabilistic risk analysis (PRA) section, of the CPPU LTR, as revised by GENE responses to the staff requests for additional information dated December 3, 2001, and December 18, 2001.

Though license applications that use the CPPU LTR will not necessarily be risk-informed submittals, the LTR does require the presentation and discussion of the plant-specific risk impacts associated with a CPPU. The staff will use the guidelines delineated in RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," to focus their evaluation of these plant-specific impacts. This evaluation will include a review of the CPPU impacts to core damage frequency (CDF) and large early release frequency (LERF) due to internal events, external events (e.g., fire, seismic, and high winds), and shutdown operations. The staff evaluation will also address the quality of the licensee's plant-specific PRA that is used to support the licensee's analyses and conclusion that the risk impacts associated with a CPPU are acceptably small.

Each of the major PRA-related review areas presented in the CPPU LTR, as modified by the GENE responses to the staff's requests for additional information, are discussed in the following subsections.

10.4.1 Internal Events

The CPPU LTR PRA section discusses the potential effect of CPPU from internal events, specifically identifying the need for licensees in plant-specific submittals, to address the risk impacts of CPPU associated with initiating event frequency, component reliability, operator response, and success criteria. The staff agrees that the licensee's plant-specific submittal should individually address the risk impacts of CPPU in each of these areas of the internal events analysis, as well as address the overall risk impact of CPPU for internal events. To determine that the risk impacts of CPPU from internal events are acceptable, the staff will evaluate the information provided by the licensee in plant-specific submittals using the guidance provided in RG 1.174. Therefore, the licensee should provide and describe the change in risk associated with implementing the CPPU for each of these areas of the internal events analysis and demonstrate that the overall change in risk from internal events is acceptably small. If the licensee's analysis identifies any vulnerabilities or took credit for plant modifications that had not been implemented when the analysis was conducted, the licensee should identify these conditions, resolution of these conditions for CPPU conditions, and demonstrate, either quantitatively or qualitatively, that these risk impacts of CPPU are acceptably small. If the estimated change in CDF and/or LERF, or base CDF and/or LERF, exceeds the RG 1.174 guidelines, including the consideration of the existence of a potential vulnerability or if a potential vulnerability is introduced by the CPPU, the licensee should provide a more detailed justification to support the acceptability of implementing the CPPU. The licensee's information needs to be sufficient for the staff to conclude that the risk impact of CPPU from internal events is acceptably small.

10.4.2 External Events

The PRA section in the CPPU LTR discusses the potential effect of CPPU from external events and identifies the need for licensees, in plant-specific submittals, to address the risk impacts of

CPPU from external events. The staff agrees that the licensee's plant-specific submittal should address the risk impacts of CPPU for external events.

To determine that the risk impacts of CPPU from external events are acceptable, the staff will evaluate the information provided by licensees in plant-specific submittals using the guidance provided in RG 1.174. Therefore, if the licensee has a PRA for some external events, the change in risk associated with implementing the CPPU for these external events should be described and the submittal should demonstrate that this change in risk is acceptably small. If the licensee does not have a PRA for some external events, such as, if a margins-type analysis was performed as part of the individual plant evaluation of external events (IPEEE) for the plant, the submittal should describe how the CPPU affects these external events analyses. If the analysis identified any vulnerabilities, outliers, or anomalies or took credit for plant modifications that had not been implemented when the analysis was conducted, the licensee should identify these conditions, the resolution of these conditions for CPPU conditions, and demonstrate, either quantitatively or qualitatively, that the risk impacts of CPPU associated with these external events are acceptably small. If the estimated change in CDF and/or LERF, or base CDF and/or LERF, exceeds the RG 1.174 guidelines, including the consideration of the existence of a potential vulnerability, such as may be identified in a margins-type analysis or if new potential vulnerabilities are introduced by the CPPU, the licensee should provide a more detailed justification to support the acceptability of implementing the CPPU. The licensee's information needs to be sufficient for the staff to conclude that the risk impact of CPPU from external events is acceptably small.

10.4.3 Shutdown Risks

The CPPU LTR PRA section discusses the potential effect of CPPU from shutdown operations and identifies the need for licensees, in plant-specific submittals, to address the risk impacts of CPPU on shutdown operations. The staff agrees that the licensee's plant-specific submittal should address the risk impacts of CPPU for shutdown operations, including a description of how the licensee manages and controls risks during shutdown operations.

To determine that the impacts of CPPU on shutdown risk are acceptable, the staff will evaluate the information provided by the licensee in plant-specific submittals using the guidance provided in RG 1.174. Therefore, if the licensee has a shutdown PRA, the change in risk associated with implementing the CPPU should be described and the submittal should demonstrate that this change in risk is acceptably small. If the licensee does not have a shutdown PRA, the submittal should discuss how the CPPU affects shutdown risks, how the risks are managed, and address any critical or time-limited conditions to demonstrate that these risks are acceptably small and properly controlled at CPPU conditions. If the estimated change in CDF and/or LERF, or base CDF and/or LERF, exceeds the RG 1.174 guidelines, including the consideration of potential vulnerabilities, weaknesses, or limitations in the licensee's shutdown risk management approach or if new potential vulnerabilities are introduced by the CPPU, the licensee should provide a more detailed justification to support the acceptability of implementing the CPPU. The licensee's information needs to be sufficient for the staff to conclude that the risk impact of CPPU for shutdown operations is acceptably small.

10.4.4 PRA Quality

The CPPU LTR PRA section discusses the need for licensees to address the quality of the supporting PRA. The staff agrees that the licensee's plant-specific submittal should address the scope, level of detail, and quality of the PRA and other relied upon risk-related evaluations (e.g., seismic margins analysis) used to support the licensee's determination that the risk impacts associated with CPPU are acceptably small for internal events, external events, and shutdown operations.

The scope, level of detail, and quality of the licensee's plant-specific PRA and other risk-related evaluations used to support a CPPU license application should be commensurate with the application for which it is intended and the role that the PRA results play in the utility's and staff's decision-making process and should be commensurate with the degree of rigor needed to provide a valid technical basis for the staff's decision. To determine that the PRA used in support of the license application is of sufficient quality, scope, and level of detail, the staff will evaluate the information provided by the licensee in plant-specific submittals using the guidance provided in RG 1.174, as well as consider the staff's previous reviews on the licensee's individual plant examination (IPE) and IPEEE submittals and the conclusions and findings of any industry or independent peer reviews. Therefore, the licensee should discuss how the process used to update and maintain the PRA and supporting analyses (e.g., thermal hydraulic calculations) would ensure that they are representative of the as-built, as-operated plant and should discuss any previously identified weaknesses, review findings, and conclusions on the IPE and IPEEE, up through the current PRA model, to ensure that they have been adequately considered and addressed in the CPPU analyses. The licensee's information needs to be sufficient for the staff to conclude that the PRA and other relied upon risk-related evaluations adequately reflect the as-built, as-operated plant for the specific CPPU license application.

10.4.5 Conclusions

Based on the staff's review of the PRA section of the CPPU LTR, as modified by the GENE responses to the staff's requests for additional information, the staff finds this section acceptable for use by licensees as the overall, high-level guidance for the plant-specific submittals of PRA information in support of CPPU license applications. However, the staff notes that in this section of the CPPU LTR, evaluations were dispositioned as being completely plant-specific and the staff recognizes that GENE has created a submittal shell report, based on their experience with prior extended power uprate reviews, to aid licensees in providing the proper level of detail and quality of information to address the plant-specific risks associated with a CPPU. The staff expects GENE to continue to use and improve upon this shell report to support the licensee's plant-specific submittals by proactively addressing the risk-related issues and topics that have previously been raised by the staff for extended power uprates.

The staff further notes that it will use the guidelines delineated in RG 1.174 to focus its evaluation of the plant-specific risk impacts associated with a licensee's CPPU license application. This evaluation will include a review of the CPPU impacts to CDF and LERF due to internal events, external events, and shutdown operations. The staff evaluation will also address the quality of the licensee's plant-specific PRA and other relied upon risk-related evaluations that are used to support the licensee's analyses and conclusion that the risk impacts associated with a CPPU are acceptable.

10.5 Testing

The "Testing" section of the CPPU LTR states that testing is required for the initial power ascension following the implementation of CPPU, [
.] The CPPU LTR establishes a standard set of tests for the initial power

ascension steps of CPPU. [

.]

10.5.1 Testing Program

The following power increase-testing, which supplements normal TS testing, is provided in the CPPU LTR:

- Testing will be done in accordance with TS surveillance requirements on instrumentation that is re-calibrated for CPPU conditions. Overlap between the intermediate range monitor and the APRM will be assured.
- Steady-state data will be taken at points from 90 percent up to 100 percent of the pre-uprated thermal power so that system performance parameters can be projected for CPPU conditions before the pre-uprated power rating is exceeded.
- Power increases will be made along an established flow control/rod line in increments of less than or equal to 5 percent 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 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 power increment, 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 test unless they have been replaced by updated criteria since the initial test program. [

.]

10.5.2 Large Transient Tests

In the CPPU LTR, GENE proposed that large transient tests (MSIV closure and generator load rejection tests similar to those conducted during initial plant startup) included in the NRC-approved topical report ELTR1 not be performed for CPPU. ELTR1 includes the MSIV closure test for power uprates greater than 10 percent above any previously recorded MSIV closure transient data and the generator load rejection test for power uprates greater than 15 percent above any previously recorded generator load rejection transient data. GENE provided justification for not performing these tests and concluded that they are not needed to demonstrate the safety of plants implementing a CPPU.

In evaluating GENE's justification not to perform the two large transient tests, the staff considered: (1) the modifications made to the plant for a CPPU that are related to the two tests, (2) component and system level testing that will be performed either as part of the licensee's power ascension and test plan or to meet TS surveillance requirements, (3) past experience at other plants, and (4) the importance of the additional information that could be obtained from performing the two tests with respect to plant analyses.

10.5.3 Transient Tests and Modifications

Large transient testing is normally performed on new plants because experience does not exist to confirm a plant's operation and response to events. However, these tests are not normally performed for plant modifications following initial startup because of well-established quality assurance and maintenance programs including component and system level post- modification testing and extensive experience with general behavior of unmodified equipment. When major modifications are made to the plant, large transient testing may be needed to confirm that the modifications were correctly implemented. However, such testing should only be imposed if it is deemed necessary to demonstrate safe operation of the plant.

GENE stated that large transient tests only challenge a limited set of systems and components and provided information regarding such systems. This situation results because the scram rapidly reduces power and the long-term consequences are relatively benign and controlled by normal operator actions. For example, the system requirements for the required actions for the closure of all MSIVs and load rejection are limited to the following:

Closure of all MSIVs	Systems Challenged
[	
	]

Load Rejection	Systems Challenged
[	
	]

The instrumentation that initiates these actions is also included in the list of required functions. The only other components that have a significant effect on these transients are [

.] Since feedwater flow for normal operation is significantly greater than that required to restore water inventory and for decay heat removal following a scram, a large transient test is not required.

[

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10.5.4 Other Testing

Regarding the testing requirements for the required systems other than feedwater, GENE provided a list of applicable requirements in the Standard Technical Specifications (STS) that are representative of a typical plant's testing requirements. The feedwater system is a normal operating system and its normal operating requirements are more demanding than requirements for these transient tests. The STS system/component test requirements are:

System	Technical Specification Surveillance
[	
	]

GENE concluded that meeting these TS requirements is sufficient to demonstrate the system and/or component initiation setpoint and performance characteristics. [

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10.5.5 Experience

GENE provided information on testing and or events that occurred at previously uprated BWR plants. Tests were performed at a foreign plant to demonstrate modifications made to its system to enable it to successfully avoid a scram during a turbine trip or generator load rejection transient from full power. These tests involved turbine trips at 110.5 percent OLTP and 113.5 percent OLTP and a generator load rejection test at 104.2 percent OLTP. The testing demonstrated the performance of equipment that was modified in preparation for higher power levels. Equipment that was not modified performed as before. The reactor vessel pressure was controlled at the same operating point for all the tests. No unexpected performance was observed except in the fine-tuning of the turbine bypass opening that was done as the series of tests progressed. These large transient tests at the foreign plant demonstrated the response of the equipment and the reactor response. The observed response closely matched the predicted response from which GENE concluded that the licensing analyses for uprated conditions reflected the behavior of the plant.

The three unplanned transients at BWR plants included two load rejections and a turbine trip subsequent to power uprate. In each case the licensee concluded that no anomalies were seen in the plant's response to these events, and the behavior of the primary mitigation systems was as expected. No new plant behavior was observed.

10.5.6 Safety Analyses and Mitigation Capability

GENE maintains that the database on large transient testing is extensive. All plants performed a rapid pressurization transient test at essentially the OLTP during initial startup testing. The purpose of the test was primarily to demonstrate the installed systems' mitigation capability. In addition, it provided a substantial amount of information used for transient model qualification for the specific plant systems' performance. To further support transient model qualification, separate transient testing was performed. These tests were designed to capture the important transient effects. These tests provide the primary test data used in the development of all current BWR transient analysis models used for the simulation of rapid pressurization transients such as the MSIV closure and generator load rejection.

[

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10.5.7 New Systems or Features

GENE has concluded that current power uprate experience has demonstrated that new systems or features are not required to mitigate the consequences of rapid pressurization transients. Therefore, for the typical CPPU, there is no need to perform large transient testing to test new systems or features. [.]
In addition, GENE has committed that, if a new system or feature is required to mitigate a rapid pressurization event for CPPU, its need will be documented in the plant-specific PUSAR. [

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10.5.8 Staff Evaluation

Section 50.92, "Issuance of amendment" states, in part, that for the determination of issuing amendments, the Commission will be guided by the considerations which govern the issuance of initial licenses or construction permits to the extent applicable and appropriate.

Section 50.34, "Contents of Applications: Technical Information" addresses initial licenses. It requires, in part, that a licensee include the principal design criteria in the safety analysis. The Introduction to 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," states that these principal design criteria are to establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety.

Regarding testing and as stated in RG 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants," the primary objectives of a suitable test program are to:

(1) Provide additional assurance that the facility has been adequately designed and, to the extent practical, to validate the analytical models and to verify the correctness or conservatism of assumptions used for predicting plant responses to anticipated transients and postulated accidents, and

(2) Provide assurance that construction and installation of equipment in the facility have been accomplished in accordance with design. The staff based its acceptance of the CPPU testing program on these objectives.

The staff is developing guidance to generically address the requirement for conducting large transient tests in conjunction with power uprates. Therefore, the staff is not prepared to accept the proposed elimination of large transient tests for CPPU. The staff intends to issue a supplement to this safety evaluation when the guidance is available. As part of the PUSAR, the plants will address this subject.

10.5.9 Conclusion

ELTR1 has been accepted by the NRC as the review basis for EPU amendment requests. The CPPU LTR also includes [ ] guidelines for testing, but has eliminated the recommendation in ELTR1 to perform large transient tests. The staff has previously accepted not performing these tests on a plant-specific basis. However, the staff is developing guidance to generically address the requirement for conducting large transient tests in conjunction with power uprates. Therefore, the staff is not prepared at this time to accept the proposed elimination of large transient tests for CPPU.

The staff finds that the performance of numerous component, system, and other testing, in combination with the evaluation of the systems and components and operating experience discussed above, is sufficient to satisfactorily demonstrate successful plant modifications and performance. The staff also finds that information obtained from the MSIV closure and generator load rejection tests could be useful to confirm plant performance, adjust plant control systems, and enhance training material. However, the staff will consider, on a plant-specific basis, the need to conduct these tests (i.e., the risk due to potential random equipment failures during the test); and the additional burden that would be imposed on the licensee.

The staff concludes that the GENE test program, with the exception of the proposal to eliminate large transient testing (i.e., MSIV closure and turbine generator load rejection), meets the objectives of a suitable test program in that the testing included in the program provides additional assurance that the CPPU design is adequate and it provides assurance that the modifications and installation of equipment as part of a CPPU is accomplished in accordance with design.

10.6 Human Factors Evaluation

The staff reviewed the CPPU LTR and the additional information provided by letter dated December 3, 2001, in the area of human factors and operator response considerations.

10.6.1 <u>Operator Response</u>

[

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Based on PRA experience for uprated BWRs, some effect is expected on PRA results such as CDF and LERF. The CPPU effect will be determined when the plant-specific PRA is revised. GENE has dispositioned operator response as plant-specific. In their December 3, 2001, letter, GENE provided additional information stating that GENE will update the CPPU PUSAR shell document to indicate that the expected level of detail for a plant-specific submittal would be as follows:

- Explain and justify any changes in plant risk that result from changes in risk-important operator actions.
- Describe any new risk-important operator actions required as a result of the proposed power uprate and changes (e.g., reduced time available or additional time required) to any current risk-important operator actions that will occur as a result of the power uprate.
- Describe the specific procedural steps involved in these actions.
- Address any operator workarounds that might affect these response times.
- Identify any operator actions that are being automated as a result of the power uprate.

The staff concludes that the effect of CPPU on operator response and plant risk is plant-specific. Licensees applying for CPPU should provide plant-specific information as described above and in the GENE PUSAR shell.

10.6.2 <u>Operator Training</u>

The CPPU LTR states that classroom training will address "various aspects of CPPU" and provides examples. [.] In Attachment 1 to their December 3, 2001 letter, GENE provided additional information to clarify that the examples cited in the LTR are provided for information only and not as a plant-specific commitment. GENE also noted that changes to operator training are considered as part of the

CPPU implementation plan, are expected to be limited in scope, will be made consistent with current plant training program requirements, and must be completed and implemented prior to any power ascension above the currently licensed power level.

The LTR states that simulator changes and fidelity revalidation will be performed in accordance with ANSI/ANS 3.5-1985. By letter dated December 3, 2001, GENE committed to revise the LTR to read:

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

GENE also noted that simulator changes and fidelity revalidation are considered as part of the CPPU implementation plan and that any required simulator changes and revalidation will be completed and implemented prior to any power ascension above the pre-CPPU power level. The staff concludes that the CPPU LTR, as revised by GENE's December 3, 2001, letter, describes controls providing adequate assurance for the training of operators for CPPU.

10.6.3 Human Factors

In the CPPU LTR, GENE dispositioned human factors changes that may be necessary to support CPPU as a plant-specific matter. By letter dated December 3, 2001, GENE provided additional information stating that human factors changes depend on the specific plant modifications, are the responsibility of the licensee, and will be made consistent with current program requirements. GENE also stated that human factors changes are considered as part of the CPPU implementation plan and any required changes will be completed and implemented prior to any power ascension above the currently licensed power level. The staff concludes that the effect of CPPU on control room human factors is plant-specific. Licensees applying for constant pressure power uprate should provide plant-specific information describing the human factors changes resulting from CPPU.

10.6.4 Emergency and Abnormal Operating Procedures

The CPPU LTR states that operator actions in the emergency operating procedures (EOPs) are not changed as a result of increasing rated reactor power, only the conditions at which some of the actions specified will change. The report also states that EOP curves and limits may also be included in the safety parameter display system and will be updated accordingly. By letter dated December 3, 2001, GENE stated that the LTR will be revised to note that abnormal operating procedures (AOPS) include event based operator actions and that some of the actions may be influenced by plant modifications required to support the increase in rated reactor power. GENE also stated that operating procedure changes will be made consistent with the plant requirements for their updating. In addition, changes to operating procedures are considered as part of the CPPU implementation plan and that these changes must be completed and implemented prior to any power ascension above the currently licensed power level.

Consistent with the CPPU LTR, licensees applying for a CPPU will provide plant-specific descriptions of the effect of CPPU on the plant's EOPs and AOPs.

10.7 Plant Life

In the report, GENE states that two degradation mechanisms are affected by CPPU: (1) irradiation assisted stress corrosion cracking (IASCC), and (2) flow accelerated corrosion (FAC). GENE stated that the potential increase in irradiation of the core internal components may influence IASCC, and that the increase in steam and feedwater flow rates influences FAC.

10.7.1 Irradiation Assisted Stress Corrosion Cracking

With regard to increasing the potential of the core internals to be affected by IASCC, GENE states that the peak fluence increase experienced by the reactor internals does not represent a significant increase in the potential for IASCC, and therefore the current inspection strategy for the reactor internal components is expected to be adequate to manage any potential effects of the power uprate on the integrity of the components.

In its RAI, the staff informed GENE that operating experience has identified stress corrosion cracking as a mechanism active in both domestic and foreign BWR plants and as a result, the BWR owners have established the BWR Vessel and Internals Project (BWRVIP), which mandated an augmented inspection program for the reactor internals. The BWRVIP program has been reviewed and approved by the staff as being adequate to control and manage degradation of BWR safety-related reactor internals. The staff requested that GENE address this issue by specifying in its LTR that each licensee applying for power uprate should implement the BWRVIP recommendations. By letter dated December 18, 2001, GENE provided revised wording for Section 10.7 of the CPPU LTR that directed the licensees to conform to the staff's request. The staff concludes that GENE's revised wording concerning the reactor vessel internals in Section 10.7 of the CPPU LTR provides enough specific direction to a licensee for assessing the impact of a proposed CPPU on the susceptibility of the reactor internal components to IASCC.

10.7.2 Flow Accelerated Corrosion

FAC occurs in carbon steel components exposed to flowing single or two-phase water. These components may be located in NSSS, turbine generator, or BOP. Therefore, all plants are recommended to have adequate FAC programs for managing any potential effects of FAC in these systems. FAC depends on several plant parameters, including fluid flow velocity in subject piping. Increased fluid flow velocity will increase FAC. A power uprate of 20 percent will cause higher steam flow and the corresponding flow of feedwater and will influence FAC in the susceptible components. Although this influence in many cases may not be very significant, the phenomenon is plant-specific, and in order to determine its significance, a plant-specific evaluation should be performed. In the response to the staff's request to provide more consistent guidance for preparation of plant-specific submittals, GENE will add to the CPPU LTR a requirement that the plant uprate reports contain descriptions of plant-specific FAC programs and methodologies and specify how the programs will be adjusted as a result of power uprate. The staff has reviewed the GENE guidance for evaluating effects of the power uprate on FAC and finds it to be satisfactory.

11.0 CPPU CONDITIONS AND LIMITATIONS

11.1 General Comment on Use of CPPU LTR

Licensees proposing to reference the CPPU LTR 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 mentioned in the seven restrictions applicable to the CPPU LTR must first request and obtain a license amendment for the associated change in accordance with the CPPU LTR. The one exception is with regards to a source term methodology change. A licensee may submit and the NRC staff will review a source term methodology change, in lieu of the analysis in Section 9.2 of the CPPU LTR, concurrent with the power uprate request, if the source term submittal supports operation at the uprated power level. Licensees proposing to utilize fuel designs other than GE fuel, up through GE 14 fuel, may not reference the CPPU LTR as a basis for their power uprate since the CPPU LTR process applies only to GE fuel and GE accident analysis methods. However, such licensees may reference the CPPU LTR for areas other than those involving reactor systems and fuel issues which are not impacted by the fuel design. Licensees should afford the staff sufficient time to complete its review of all associated licensing basis changes prior to submittal or request for the implementation of the power uprate when referencing the CPPU LTR.

11.2 Application of the CPPU LTR

Each of the sections of the CPPU LTR were in one of two disposition categories:

- Generic assessment
- Plant-specific evaluation

The generic assessments are those safety evaluations that can be dispositioned for a group or for 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.

Licensees using this LTR must provide analyses demonstrating the generic analyses bound their plant. If not, a plant-specific analyses must be provided or provide the plant-specific analyses, as required by each section of this safety evaluation.

12.0 REFERENCES

1. Letter from J. F. Klapproth (GE Nuclear Energy), to U.S. Nuclear Regulatory Commission, "Submittal of Proprietary Licensing Topical Report NEDC-33004P, 'Constant Pressure Power Uprate'," March 19, 2001.

2. Letter from J. F. Klapproth (GE Nuclear Energy), to U.S. Nuclear Regulatory Commission, "Submittal of Proprietary Licensing Topical Report NEDC-33004P, 'Constant Pressure Power Uprate', Revision 1," July 26, 2001.

3. Letter from J. F. Klapproth (GE Nuclear Energy), to U.S. Nuclear Regulatory Commission, "Responses to Draft Request for Additional Information (RAI) to Licensing Topical Report NEDC-33004P, Revision 1," December 3, 2001.

4. Letter from J. F. Klapproth (GE Nuclear Energy), to U.S. Nuclear Regulatory Commission, "Responses to Draft Request for Additional Information (RAI) to Licensing Topical Report NEDC-33004P, Revision 1," December 18, 2001.

5. Letter from J. F. Klapproth (GE Nuclear Energy), to U.S. Nuclear Regulatory Commission, "Responses to Request for Additional Information (RAI) Set Number 9 from RXSB, Questions 10 through 14 and Set Number 10, Large Transient Testing Pertaining to Licensing Topical Report NEDC-33004P, Revision 1," December 21, 2001.

6. Letter from J. F. Klapproth (GE Nuclear Energy), to U.S. Nuclear Regulatory Commission, "Errata & Addenda 1, Update of GE Proprietary Licensing Topical Report NEDC-33004P, 'Constant Pressure Power Uprate', Revision 1," December 21, 2001.

7. GE Nuclear Energy, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," NEDC-32424P-A, February 1999 [Known as ELTR1].

8. GE Nuclear Energy, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate," NEDC-32523P-A, February 2000, Supplement 1, Volume I, February 1999, and Supplement 1, Volume II, April, 1999 [Known as ELTR2].

9. Letter from Thomas H. Essig (NRC), to Joseph F. Quirk (GENE), "Staff Safety Evaluation of General Electric Boiling Water Reactor (BWR) Extended Power Uprate Generic Analyses," September 14, 1998.

10. Letter from Dennis M. Crutchfield (NRC), to G. L. Sozzi (GENE), "Staff Position Concerning General Electric Boiling Water Reactor Extended Power Uprate Program," February 8, 1996.

11. GE Nuclear Energy, "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A and NEDE-24011-P-A-US, (latest approved revision), [known as GESTAR-II].

12. GE Nuclear Energy, "10 CFR 50 Appendix G Equivalent Margin Analysis for Low Upper Shelf Energy in BWR/2 Through BWR/6 Vessel," NEDO-32205-A, Revision 1, February 1994.

13. GE Services Information Letter (SIL) No. 377, "RCIC Startup Transient Improvement with Steam Bypass."

14. NEDC-31336, "General Electric Instrumentation Setpoint Methodology."

15. NEDO-24222, "Assessment of BWR Mitigation of ATWS", Volume II, December 1979.

21. Letter from J. F. Klapproth (GE Nuclear Energy), to U.S. Nuclear Regulatory Commission, "Submittal of Proprietary Licensing Topical Report NEDC-33004P, 'Constant Pressure Power Uprate', errata and Addenda 2, October 7, 2002.

17. Letter from J. F. Klapproth (GE Nuclear Energy), to U.S. Nuclear Regulatory Commission, "Submittal of Proprietary Licensing Topical Report NEDC-33004P, 'Constant Pressure Power Uprate', Revision 3," February 6, 2003.

18. Letter to G. L. Sozzi (GE) from A. Thadani (NRC), "Use of the SHEX Computer Program and ANSI/ANS 5.1-1979 Decay Heat Source Term for Containment Long-Term Pressure and Temperature Analysis," July 13, 1993.

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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.

Revision 4 is the acceptance version of Revision 3. The NRC Safety Evaluation has been included in the front of the document and the RAIs and corresponding responses have been added as Appendix A. In addition, a change committed in the response to RAI Set 6 Number 2 was added to Section 8.1, and a Reference corrected in Section 7.3.

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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

[[

]]

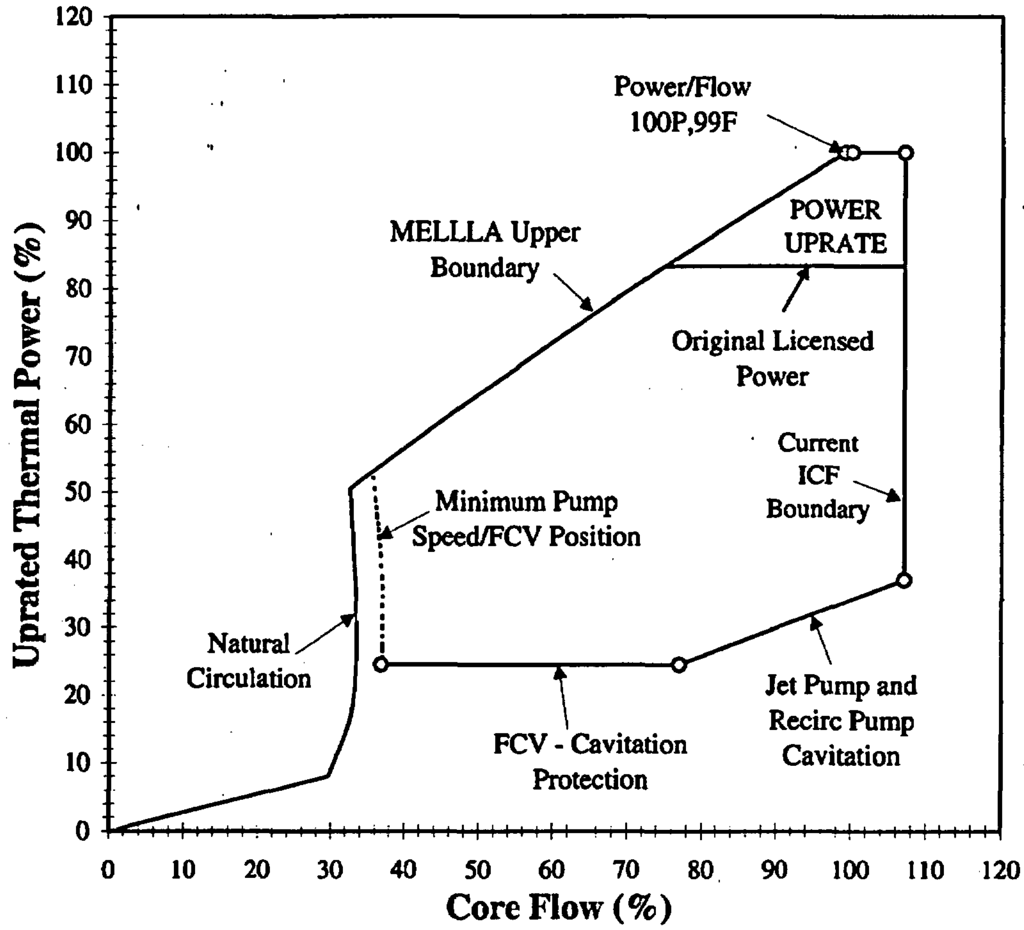

Figure 1-1. Typical CPPU-Based Power Uprate Power/Flow Map

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 [[ ]] 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	
5.3.6 RCIS Rod Withdrawal Limiter High Power Setpoint - Setpoint Calculation	Increased reactor power level and turbine first stage pressure change	

Topic	CPPU Effect	Disposition
Methodology		
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 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 (Reference 3).

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.

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

[[

]] 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

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

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