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MFN 02-013

March 14, 2002

U.S Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555-0001

Attention: Chief, Information Management Branch
Program Management
Policy Development and Analysis Staff

Subject: **Submittal of GE Non-Proprietary Licensing Topical Report NEDO-33004, Revision 1, "Constant Pressure Power Uprate"**

Reference: MFN –01-045, JF Klapproth to NRC, *Submittal of GE Non-Proprietary Licensing Topical Report NEDO-33004, Revision 1, "Constant Pressure Power Uprate"*, dated August 27, 2001

GE submitted the reference non-proprietary Constant Pressure Power Uprate report, on August 27, 2001. After discussions with the NRC staff, GE has revised the subject report and included additional information about the topics contained in the Constant Pressure Power Uprate review process.

If you have any questions about the information provided here please contact Ralph Hayes at (408) 925-5424, or myself.

Sincerely,

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Attachment: *NEDO-33004, Revision 1, "Constant Pressure Power Uprate", Class I, March 2002*

cc: JE Donoghue – USNRC
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175 Curtner Avenue
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NEDO-33004
DRF A22-00116-00
Revision 1
Class I
March 2002

LICENSING TOPICAL REPORT

CONSTANT PRESSURE POWER UPRATE

Approved: K. S. Cole
K. S. Cole, Manager
BWR Asset Enhancement Services

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 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.

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ABSTRACT

GE has previously developed and implemented an approach called Extended Power Uprate, to increase the power of operating Boiling Water Reactors (BWR) up to 120% of the present power level. 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.

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 previous experience with Extended Power Uprate projects.

To ease future NRC reviews, a prescribed approach to be used for each plant specific power uprate submittal is provided. Future plant specific submittals of Constant Pressure Power Uprate will include assessments based on the approach prescribed herein.

REVISIONS

NEDO-33004, Revision 0 was submitted for NRC review on August 7, 2001. Feedback from the NRC staff review, has been factored into this revision of NEDO-33004, Revision 1. The key changes in Revision 1 are the addition of information describing the topics included for review in Sections 2 through 10 and relocating the acronym list for consistency with NEDC-33004P. NEDO-33004, Revision 1 replaces NEDO-33004, Revision 0 in its entirety.

ACRONYMS

Acronym	Definition
ATWS	Anticipated Transient Without Scram
BOP	Balance of Plant
BWR	Boiling Water Reactor
CFR	Code of Federal Regulations
CPPU	Constant Pressure Power Uprate
CRD	Control Rod Drive
CS	Core Spray
DC	Direct Current
ECCS	Emergency Core Cooling System
ELTR 1	NEDC-32424P-A (Reference 1)
ELTR 2	NEDC-32523P-A (Reference 2)
EPU	Extended Power Uprate
GNF	Global Nuclear Fuel
HPCI	High Pressure Coolant Injection
HPCS	High Pressure Core Spray
IEEE	Institute of Electrical and Electronic Engineers
LOCA	Loss-of-Coolant Accident
LPCI	Low Pressure Coolant Injection
LPCS	Low Pressure Coolant Spray
LTR	Licensing Topical Report
MAPLHGR	Maximum Average Planer Linear Heat Generation Rate
MELLLA	Maximum Extended Load Line Limit Analysis
MEOD	Maximum Extended Operating Domain
NEMA	National Electric Manufactures' Association
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OLTP	Original Licensed Thermal Power

Acronym	Definition
PCS	Pressure Control System
PUSAR	Power Uprate Safety Analysis Report
RCIC	Reactor Core Isolation Cooling
RIPDs	Reactor Internal Pressure Differences
RTP	Rated Thermal Power
SAR	Safety Analysis Report
SBO	Station Blackout
SLCS	Standby Liquid Control System
SPU	Stretch Power Uprate
UFSAR	Updated Final Safety Analysis Report

1.0 INTRODUCTION

Previously, General Electric (GE) submitted 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 possibility that the maximum operating reactor pressure may be increased. These guidelines and evaluations, together with associated Nuclear Regulatory Commission (NRC) 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 approval of these licensing topical reports, GE developed an approach to uprating reactor power that maintains the current plant maximum operating reactor pressure. The power uprate with no pressure increase has been utilized at several plants and will be used 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. 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 approval of this LTR is received.

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.

To 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. 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. However, for those analyses and evaluations that are generically dispositioned in this report, the plant specific PUSAR is only required to confirm the applicability of the generic dispositions for the specific plant application.

The acronym for an assessment or equipment name is typically provided with the first use of the name (a table of acronyms is provided at the beginning of this report).

1.1 REPORT APPROACH

The report sections generally correspond to those used on previous 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

The technical evaluations are contained in Sections 2 through 10. General information has been provided in Section 11 to support utility licensing documentation for the plant specific CPPU submittal. This general information is provided to assist the utility in the development of the environmental report, plant technical specification changes, and significant hazards assessment. The utility may elect to reference some or all of the information given in Section 11 in the documentation supporting the plant specific CPPU licensing submittal.

1.1.1 Generic Assessments

Generic assessments are those safety evaluations that can be applied to 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 required 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 analyzed with GE methodology. If another vendor fuel design is considered as part of the power uprate, fuel design dependent generic assessments must be separately evaluated and justified.

For those CPPU assessments having a negligible effect, current CPPU experience with 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 in Reference 3.

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.

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 the assessment methodology is identified in Reference 1, 2 or 3, these documents are referenced rather than the original report.

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 is extended up to the new 100% core power value. A typical power/flow map for the power uprate conditions is shown in Figure 1-1.

1.2.2 Nuclear and Thermal-Hydraulic Evaluations

The change in the power level will affect the plant steady-state heat balance. 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. Some of the thermal-hydraulic safety analyses can be performed on a generic basis; the remaining thermal-hydraulic safety analyses require plant specific evaluations. The plant specific evaluation or confirmation of applicability to the generic assessment will be provided in the plant specific submittal.

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.

1.2.4 System Evaluations

The effect of CPPU on Nuclear Steam Supply System (NSSS) and Balance Of Plant (BOP) systems is system dependent and is described in Sections 2.0 through 9.0.

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.

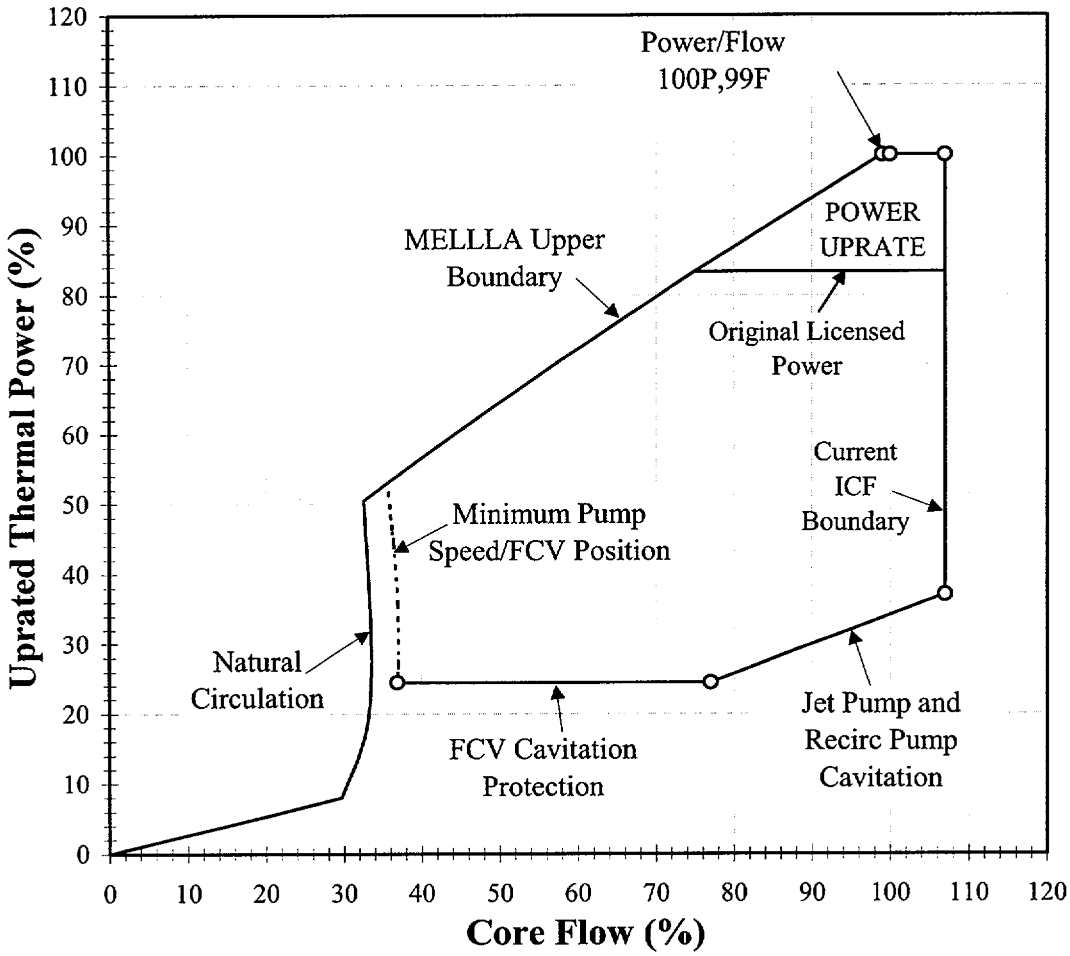

Figure 1-1. Typical CPPU-Based Power Uprate Power/Flow Map

2.0 REACTOR CORE AND FUEL PERFORMANCE

Evaluations associated with the Reactor Core and Fuel Performance addressed in this section are:

Fuel Design and Operation

All fuel and core design limits continue to be met by planned deployment of fuel enrichment and burnable poison, and supplemented by core management control rod pattern and/or core flow adjustments. Revised loading patterns, larger batch sizes, and potentially new fuel designs may be used to provide additional operating flexibility and maintain fuel cycle length.

Thermal Limits Assessment

Operating limits ensure that regulatory and/or safety limits are not exceeded for a range of postulated events [e.g., transients, loss-of-coolant accidents (LOCA)]. Cycle-specific core configurations, evaluated for each reload, confirm uprated capability and establish or confirm cycle-specific limits, as is currently the practice. The evaluation of thermal limits for the uprated core shows that the thermal margin design limits are acceptable.

Reactivity Characteristics

All minimum shutdown margin requirements apply to cold conditions and are maintained without change. The Technical Specifications cold shutdown margin requirements are not affected. Operation at higher power could reduce the hot excess reactivity during the cycle. This loss of reactivity does not affect safety, and is not expected to significantly affect the ability to manage the power distribution through the cycle to achieve the target power level.

The CPPU power-flow operating map (Figure 1-1) includes the operating domain for the CPPU. The maximum thermal operating power and maximum core flow shown on Figure 1-1, correspond to the CPPU rated thermal power (RTP).

Thermal Hydraulic Stability

Four long-term stability options: Enhanced Option I-A, Option I-D, Option II, and Option III have been developed for all GE BWR product lines. For CPPU, the long-term stability options are evaluated for operation at the CPPU conditions. The stability interim corrective actions applicability is evaluated when applicable.

Reactivity Control

The Control Rod Drive (CRD) system changes the core reactivity by positioning neutron absorbing control rods within the reactor. It is also required to scram the reactor by rapidly inserting withdrawn rods into the core.

The components of the CRD mechanism, forming part of the primary pressure boundary, have been designed in accordance with the ASME Boiler and Pressure Vessel (B&PV) Code, Section III. The CPPU conditions are evaluated to ensure that limiting CRD component stresses are within the allowable stress criteria and that the current fatigue analysis is valid.

3.0 REACTOR COOLANT AND CONNECTED SYSTEMS

Evaluations associated with the Reactor Coolant and Connected Systems addressed in this section are:

Nuclear System Pressure Relief/Overpressure Protection

The purpose of the nuclear system pressure relief is to prevent overpressurization of the nuclear system during abnormal operational transients. The safety relief valves (SRVs) along with reactor scram provide this protection. The SRV setpoints are not changed with CPPU, because the maximum operating dome pressure is not changed.

The design pressure of the reactor vessel and reactor pressure coolant boundary remains the same with CPPU. The acceptance limit for pressurization events is the ASME code allowable peak pressure of 110% of the current design value. The limiting pressurization event is not changed by CPPU .

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 is evaluated for CPPU operation at the higher steam flow rates.

Reactor Vessel

The reactor vessel wall fluence analyses will be performed consistent with NRC-approved methods.

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. These components are evaluated for CPPU.

Reactor Internals

The increase in core average power results in higher core loads and reactor internal pressure differences (RIPDs) due to the higher core exit steam quality. The RIPDs are evaluated for normal steady-state operation, upset, and faulted conditions for all major reactor internal components.

A reactor internals structural evaluation of the key reactor internal components is performed to assess the structural integrity for the load changes associated with CPPU. This evaluation is used to demonstrate that the structural integrity of the core support and non-core support structure reactor internal components is maintained in the CPPU operating condition, consistent with the design basis.

Flow-Induced Vibration

The Main Steam (MS) and Feedwater (FW) systems experience increases in flow due to CPPU. The MS and FW piping systems (inside containment) are evaluated for the increases in related loads to ensure these load changes do not result in load limits being exceeded for the MS or FW piping system or for interfacing RPV nozzles, penetrations, flanges or valves.

A flow-induced vibration evaluation is used to demonstrate that operation at CPPU power and flow conditions is possible without any detrimental effects on the safety-related reactor internal components.

Piping Evaluation

The Reactor Coolant Pressure Boundary (RCPB) Piping systems consist of a number of safety related piping subsystems that move fluid through the reactor and other safety systems

The effects of CPPU are evaluated for the reactor coolant piping systems that are part of the primary RCPB and could be affected by a CPPU-related increase in flow or operating temperature. These evaluations are used to demonstrate that CPPU does not have an adverse effect on the primary piping systems design.

The balance-of-plant (BOP) large bore and small bore ASME Section III, Class 1, 2, and 3 piping and supports are evaluated for acceptability at the CPPU conditions. The evaluation of the BOP piping and supports is performed in a manner similar to the evaluation of RCPB piping systems and supports.

Reactor Recirculation System

An evaluation of the reactor recirculation system performance at CPPU conditions is performed to determine that adequate core flow can be maintained and that CPPU power operation is within the capability of the reactor recirculation system.

Main Steamline Flow Restrictors

The main steam line flow restrictors are evaluated to ensure that the existing design margin of the flow restrictors is maintained with the changes in conditions resulting from CPPU.

Main Steam Isolation Valves

The Main Steam Isolation Valves (MSIVs) are part of the RCPB, and perform the safety function of steam line isolation during certain abnormal events. The MSIVs are evaluated for the effects of the potential effects of CPPU related changes to the safety functions of the MSIVs.

Reactor Core Isolation Cooling/Isolation Condenser

The Reactor Core Isolation Cooling (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. The RCIC system is evaluated for CPPU to ensure there are no changes to the RCIC high-pressure injection process parameters and no change to the overspeed trip margins.

The Isolation Condenser (IC) system provides core cooling in the event of a transient where the reactor pressure vessel is isolated from the main condenser concurrent with the loss of all feedwater flow. The limiting acceptance criterion for the loss of feedwater flow transient event is to provide adequate core cooling during the transient by maintaining sufficient water level inside the core shroud to ensure that the top of active fuel remains covered throughout the event. The IC system at CPPU conditions is evaluated to ensure that CPPU does not affect the performance capability of the system, and does not exceed any of the original design pressures or temperatures for the system components.

Residual Heat Removal System

The RHR system is designed to restore and maintain the coolant inventory in the reactor vessel and to remove sensible and decay heat from the primary system and containment following reactor shutdown for both normal and post accident conditions. The RHR system equipment and operating modes are evaluated to ensure that CPPU does not prevent any of the RHR modes from performing their intended functions.

Reactor Water Cleanup System

The Reactor Water Cleanup (RWCU) system is designed to remove solid and dissolved impurities from reactor coolant, thereby reducing the concentration of radioactive and corrosive species. RWCU system operation at the CPPU conditions is evaluated to ensure the reactor water conductivity limits will be met. The containment isolation function of the RWCU containment isolation valves is addressed in Section 4.

4.0 ENGINEERED SAFETY FEATURES

Evaluations associated with the Engineered Safety Features addressed in this section are:

Containment System Performance

The USAR provides the containment responses to various postulated accidents that validate the design basis for the containment. Operation during CPPU changes some of the conditions for the containment analyses. The containment pressure and temperature responses are evaluated to ensure they remain within the design limits at the CPPU conditions.

The LOCA containment dynamic loads analysis for CPPU are based primarily on the short-term main steam line break and recirculation line break LOCA analyses. The LOCA dynamic loads with the CPPU include pool swell, condensation oscillation (CO) and chugging.

The containment analyses performed for the dynamic loads evaluations will demonstrate that the short-term containment response conditions are within the range of test conditions used to define the loads for the plant. The containment response with the CPPU for times beyond the initial blowdown period are evaluated for acceptability at the CPPU conditions.

The system design for containment isolation including the capability of the actuation devices to perform with the higher flow and temperature during normal operations and under post-accident conditions is evaluated for the CPPU.

The motor-operated (MOV) requirements in the USAR and the functional requirements of the Generic Letter (GL) 89-10 MOVs are evaluated for operation at the CPPU conditions. The operability of MOVs is documented as part of the plant's GL 89-10 MOV program. If specific valves require calculation revisions, actuator adjustments and/or physical changes to ensure satisfactory performance, they are upgraded, adjusted, or modified, as necessary.

Similarly, Generic Letters 89-16, 95-07, and 96-06 are addressed to ensure continued compliance at the CPPU conditions.

Emergency Core Cooling Systems

The Emergency Core Cooling Systems (ECCS) provide protection against hypothetical LOCAs caused by ruptures in the primary system piping. The functional capability of each system is evaluated for the CPPU.

The High Pressure Coolant Injection (HPCI) or the High Pressure Core Spray (HPCS) system is evaluated to demonstrate that it can meet the design basis requirement to provide coolant flow to the reactor vessel following small breaks, and the function of fulfilling the objectives of the RCIC system in response to a transient event.

The Core Spray (CS) or Low Pressure Core Spray (LPCS) system is automatically initiated in the event of a LOCA. When operating in conjunction with other ECCS, the LPCS System is required to provide adequate core cooling for all LOCA events following depressurization. The ECCS performance is evaluated to demonstrate that the existing CS/LPCS performance capability, in conjunction with the other ECCS, is adequate to meet the post-LOCA core cooling requirement for the CPPU conditions.

The Low Pressure Coolant Injection (LPCI) mode of the RHR system is automatically initiated in the event of a LOCA. When operating in conjunction with other ECCS, the LPCI mode is required to provide adequate core cooling for all LOCA events following depressurization. The ECCS performance is evaluated to ensure that the existing LPCI mode performance capability, in conjunction with the other ECCS, is adequate to meet the post-LOCA core cooling requirement for the CPPU conditions.

The effect of CPPU on net positive suction head (NPSH) for pumps taking suction from the suppression pool is evaluated to demonstrate adequate NPSH under CPPU conditions.

The Automatic Depressurization system (ADS) uses SRVs to reduce reactor pressure following a small break LOCA and failure of the high pressure ECCS. This function allows LPCI and CS/LPCS to inject coolant into the vessel. The ADS initiation logic and capacity is evaluated to demonstrate adequacy for CPPU conditions.

Emergency Core Cooling Systems Performance

ECCS performance is evaluated to ensure the requirements of 10 CFR 50.46 and 10 CFR 50 Appendix K are satisfied.

Main Control Room Atmosphere Control System

The control room HVAC system maintains a habitable environment and ensures the operability of all the components in the control room under all operating and accident conditions. The system is designed to maintain a positive pressure within the control room envelope with respect to the adjacent areas to preclude infiltration of unconditioned air, during all the operating modes except when the system is in recirculation mode or when the system is in the maximum outside air purge mode. The performance of the Main Control Room Atmosphere Control System is evaluated at the CPPU conditions to ensure the makeup air filter trains are capable of handling the iodine loading and the decay heat as a result of the deposited radionuclides.

Standby Gas Treatment System

The Standby Gas Treatment System (SGTS) controls off-site dose rates following a postulated design basis accident by limiting the release of air-borne particulates and halogens. The design flow capacity of the system maintains the secondary containment at the required negative pressure to prevent exfiltration of air from the reactor building. The SGTS is evaluated to ensure acceptable operation at the CPPU conditions.

Main Steam Isolation Valve Leakage Control System

The Main Steam Line Isolation Valve Leakage Control system (MSIVLCS) controls the release of fission products that leak through the MSIVs following a LOCA. The leakage is directed to bleed lines aided by blowers, to maintain the pressure between the inboard and outboard isolation valves and between the outboard isolation valves and the downstream shutoff valves slightly negative with respect to atmosphere. The bleed lines pass the leakage to the reactor vessel or the SGTS. The MSIVLCS is evaluated at the CPPU conditions.

Post-LOCA Combustible Gas Control System

The Combustible Gas Control System maintains the post-LOCA concentration of oxygen or hydrogen in the containment atmosphere below the lower flammability limit. Combustible gas control is achieved using either hydrogen recombiners or nitrogen dilution. The effects of CPPU are evaluated to ensure the acceptability of operator initiation times for either system and for the quantity of stored nitrogen for plants using the nitrogen dilution method of control.

5.0 INSTRUMENTATION AND CONTROL

Evaluations associated with the Instrumentation and Control addressed in this section are:

NSSS Monitoring and Control

Increases in core thermal power and steam flow affect some instrument setpoints. The NSSS process variables and instrument setpoints that could be affected by the CPPU are evaluated. The local power range monitor detectors and the traversing incore probes are evaluated for acceptable operation at the CPPU conditions.

The Rod Block Monitor (RBM) initiates a control rod block if local power exceeds a preset limit around a selected rod during withdrawal. The RBM is required to be operable when the reactor is at a predetermined percentage of rated power. This value is evaluated at the CPPU conditions.

The function of the Rod Worth Minimizer (RWM) and the Rod Control and Information system (RCIS) is to support the operator by enforcing rod patterns until reactor power has reached appropriate levels. Adjustment to the calibration value is evaluated to maintain the setpoint for CPPU.

Balance-of-Plant Monitoring and Control

Operation of the plant at the CPPU power level is evaluated for effects on the balance-of-plant (BOP) system instrumentation and control devices. Based on CPPU operating conditions for the power conversion and auxiliary systems, process control valves and instrumentation are evaluated to have sufficient range/adjustment capability for use at the expected CPPU conditions. However, some non-safety modifications may be needed to the power conversion systems to obtain full CPPU power.

The pressure control system (PCS) responds to system disturbances related to steam pressure and flow changes to control reactor pressure within its normal operating range. The PCS consists of the pressure regulation system, turbine control valve system and steam bypass valve system. The main turbine speed/load control function is performed by the main turbine-generator Electro-Hydraulic Control (EHC) system. The turbine EHC system is reviewed for the increase in core thermal power and the associated increase in rated steam flow.

The increased steam flow for EPU requires the Turbine Control Valves (TCV) to operate under different conditions. The flow capacity of the TCVs and other characteristics are evaluated to assure that all requirements regarding interaction between the turbine generator and the NSSS are addressed.

The feedwater control system is used to maintain water level control in the reactor. The capacity of the feedwater pumps, the control system capability to control the flow and the need for any control system adjustment are evaluated for CPPU operation.

The instrument setpoints associated with primary system leak detection are evaluated for CPPU conditions. Each of the following systems is evaluated:

- Main Steam Tunnel Temperature Based Leak Detection
- RWCU System Temperature Based Leak Detection
- HPCI System Temperature Based Leak Detection
- Non-Temperature Based Leak Detection

Technical Specification Instrument Setpoints

Instrument setpoints in the Technical Specifications (TS) are established for CPPU using approved setpoint methodologies. Each setpoint is selected with sufficient margin between the actual trip setting and the value used in the safety analysis (analytical limit) to allow for instrument accuracy, calibration, and drift. Sufficient margin is also provided between the actual trip setting and the normal operating limit to preclude inadvertent initiation of the protective action.

6.0 ELECTRICAL POWER AND AUXILIARY SYSTEMS

Evaluations associated with the Electrical Power and Auxiliary Systems addressed in this section are:

AC Power

The existing off-site electrical equipment is evaluated for CPPU operation with the uprated electrical output. This evaluation includes the following components:

- Isolated phase bus duct electrical adequacy
- Isolated phase bus duct cooling
- Main transformers and the associated switchyard components
- Grid stability

The emergency diesel generator power system is evaluated for CPPU to ensure the systems have sufficient capacity to support all required loads for safe shutdown, to maintain a safe shutdown condition, and to operate the engineered safety feature equipment following postulated accidents.

DC Power

The direct current (DC) loading is evaluated for reactor power dependent load effects for the CPPU conditions.

Fuel Pool

An evaluation of the Fuel Pool Cooling and Cleanup System (FPCCS) is performed to determine its ability to handle the heat load in the spent fuel pool (SFP) for CPPU implementation. The FPCCS heat exchangers are evaluated for decay heat removal capacity. Crud activity and corrosion products associated with spent fuel are evaluated for CPPU to ensure the fuel pool water quality can be maintained by the fuel pool cleanup system. The CPPU effect on the normal radiation levels around the SFP and the effect on the design of the SFP storage racks is also evaluated.

Water Systems

Evaluations of the water systems are performed to determine the effect of the CPPU on these systems. The safety-related and non-safety-related service water system capabilities are evaluated, including the environmental effects of CPPU.

Standby Liquid Control System

The Standby Liquid Control System (SLCS) is designed to shut down the reactor from rated power condition to cold shutdown in the postulated situation that all or some of the control rods cannot be inserted. The ability of the SLCS to achieve and maintain safe shutdown is evaluated for CPPU. The performance of the SLCS during a postulated anticipated transient without scram (ATWS) is evaluated at the CPPU power level.

Power Dependent Heating, Ventilation and Air Conditioning

The HVAC systems consist mainly of heating, cooling supply, exhaust and recirculation units in the turbine building, containment building and the drywell, auxiliary building, fuel handling building, control building, and the radwaste building. These systems are evaluated for the CPPU conditions to ensure that the areas serviced can be maintained within acceptable limits.

Fire Protection

The effect of CPPU operation on the fire suppression or detection systems is evaluated. Any changes in physical plant configuration or combustible loading as a result of modifications to implement the CPPU, are evaluated in accordance with the plant modification and fire protection programs. The safe shutdown systems and equipment used to achieve and maintain cold shutdown conditions are evaluated for the CPPU conditions. The operator actions required and the time available to mitigate the consequences of a fire are evaluated.

7.0 POWER CONVERSION SYSTEMS

Evaluations associated with the Power Conversion Systems addressed in this section are:

Turbine-Generator

Turbine and generator stationary and rotating components are evaluated for increased loadings, pressure drops, thrusts, stresses, overspeed capability and other design considerations to ensure that design limits are not exceeded and that plant operation remains acceptable at the CPPU condition. In addition, valves, control systems and other support systems are evaluated.

Condenser and Steam Jet Air Ejectors

The performance of the main condensers is evaluated to ensure that condenser hotwell capacities and level instrumentation are adequate for CPPU conditions. The design capacity of the steam jet air ejectors is evaluated for CPPU conditions to ensure adequate air removal is maintained.

Turbine Steam Bypass

The steam bypass system is a normal operating system and is non-safety-related. The bypass capacity is evaluated at CPPU conditions.

Feedwater and Condensate Systems

The feedwater and condensate systems are designed to provide a reliable supply of feedwater at the temperature, pressure, quality, and flow rate required by the reactor. However, these systems do not perform a system level safety-related function.

An evaluation of these systems is performed to identify equipment changes that may be necessary.

8.0 RADWASTE AND RADIATION SOURCES

Evaluations associated with the Radwaste and Radiation Sources addressed in this section are:

Liquid and Solid Waste Management

The liquid radwaste system collects, monitors, processes, and stores radioactive waste. The concentration of activated corrosion products in liquid wastes, the volume of liquid wastes, and the volume of condensate resin generated are evaluated for CPPU to assure that the requirements of 10 CFR 20 and 10 CFR 50, Appendix I continue to be met.

Gaseous Waste Management

The Gaseous Waste Management Systems collect, control, process, store, and dispose of gaseous radioactive waste generated during normal operation and abnormal operational occurrences. The gaseous waste management systems include the offgas system and various building ventilation systems. The systems are designed to meet the requirements of 10 CFR 20 and 10 CFR 50, Appendix I.

Air inleakage evacuated from the main condenser contains non-condensable radioactive gas, normally consisting of activation gases (principally N-16, O-19 and N-13) and fission product radioactive noble gases. These non-condensable gases, along with the non-radioactive air inleakage, are continuously removed from the main condensers by the steam jet air ejectors, which discharge into the offgas system. This process stream represents the major source of radioactive gas (greater than all other sources combined) exiting the primary system.

The activity of airborne effluents released through building vents and the offgas system are evaluated for CPPU conditions to ensure adequate system capacity and environmental limits are maintained.

Radiation Sources in the Reactor Core

During power operation, the radiation sources in the core include radiation from the fission process, accumulated fission products, and neutron reactions as a secondary result of fission. The radiation sources during normal operation are evaluated for the CPPU conditions.

For post-operation evaluations, two forms of source data are applied. The first is the core gamma-ray source, which is used in shielding calculations for the core and for individual fuel bundles. The second is used for post-accident evaluations, which are performed in compliance with regulatory guidance that applies different release and transport assumptions to different fission products. The radiation sources for post operation are evaluated for the CPPU conditions.

Radiation Sources in the Reactor Coolant

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. The reactor coolant contains activated corrosion products, which are the result of metallic materials entering the water and being activated in the reactor region, and fission products resulting from the escape of minute fractions of the fission products in the fuel rods. Coolant activation sources, corrosion product concentrations, and fission products are evaluated for the CPPU conditions.

Radiation Levels

Normal operation radiation levels increase slightly and are evaluated for CPPU.

Post-operation radiation levels in most areas of the plant are expected to increase. However, individual worker exposures should be maintained within acceptable limits by the site ALARA program, which controls access to radiation areas. These procedural controls are adjusted as necessary to compensate for increased radiation levels.

The change in core inventory resulting from the CPPU is expected to increase post-accident radiation levels by no more than the percentage increase in power level. A review of areas requiring post-accident occupancy (per NUREG-0737 Item II.B) is performed to ensure that access needed for accident mitigation is not significantly affected by the CPPU.

Normal Operation Offsite Doses

For the CPPU, normal operation gaseous activity levels increase slightly. The CPPU will result in slight increases in the offsite dose from noble gases, airborne particulates, iodine, tritium or liquid effluents. In addition, offsite radiation dose caused by skyshine from the turbine is not a significant exposure pathway. The normal offsite doses are evaluated at the CPPU RTP level to ensure they remain below the limits of 10 CFR 20 and 10 CFR 50, Appendix I.

9.0 REACTOR SAFETY PERFORMANCE EVALUATIONS

Evaluations associated with the Reactor Safety Performance Evaluations addressed in this section are:

Anticipated Operational Occurrences

The plant USAR evaluates the effects of a wide range of potential plant transients. Disturbances to the plant caused by a malfunction, a single equipment failure, or an operator error are investigated according to the type of initiating event per Regulatory Guide 1.70, Chapter 15. The anticipated operational occurrences are evaluated for the CPPU conditions.

The operating critical power ratio limit is supplied in the cycle specific Core Operating Limit Report. The Technical Specification critical power ratio safety limit, thermal limits monitoring Limiting Conditions for Operation thresholds, and Surveillance Requirement thresholds are evaluated for the CPPU condition.

Design Basis Accidents

The power dependent radiological assessments reported in the UFSAR are re-evaluated for CPPU. The radiological analyses are performed based on CPPU conditions for selected postulated accidents. The events re-evaluated are the main steamline break outside containment, instrument line breaks, the Loss-of-Coolant Accident (LOCA), the Fuel Handling Accident, and the Control Rod Drop Accident. The evaluation ensures that the plant will continue to meet the applicable regulatory exposure values.

Special Events

An ATWS analysis for the CPPU condition is performed to ensure that the peak vessel pressure, peak clad temperature, peak clad oxidation, peak suppression pool temperature, and peak containment pressure meet the acceptance criteria.

The plant responses to and coping capabilities for a station blackout (SBO) event are evaluated for CPPU operation to ensure there are no changes to the systems or equipment necessary to respond to an SBO, nor is the required coping time changed, and the plant will continue to meet the requirements of 10 CFR 50.63.

The ATWS with core instability event is postulated to occur at natural circulation following a recirculation pump trip. The event is evaluated for CPPU conditions.

10.0 OTHER EVALUATIONS

Evaluations associated with Other Evaluation Topics addressed in this section are:

High Energy Line Break

High energy line breaks (HELBs) are evaluated for their effects on equipment qualification.

The HELB analysis evaluation for CPPU ensures continued support of the safety-related function and includes the following topics:

- Pipe whip and jet impingement loads and targets,
- Pipe whip restraints and jet impingement shields and their supports,
- Plant internal flooding,
- Break locations.

Moderate Energy Line Break

Moderate energy line breaks (MELB) are evaluated at the CPPU conditions to ensure that the plant flooding and environmental qualification of equipment is not affected by the CPPU for plants with MELB in their licensing basis.

Environmental Qualification

The safety-related electrical equipment, mechanical equipment with non-metallic components, and the mechanical design of equipment/components environmental qualification documentation are reviewed to assure the existing qualification remains adequate. Any changes resulting from the CPPU (radiation, pressure, temperature and humidity, as applicable) to the environmental conditions of affected safety-related equipment inside and outside containment are evaluated.

Testing

Compared to the initial startup program, CPPU requires only limited subset of the original startup test program. The same performance criteria will be used as in the original power ascension tests, unless they have been replaced by updated criteria since the initial test program. The test program includes the following:

- Technical Specification instrumentation surveillance testing,
- Control system tests,
- Steam separator-dryer performance monitoring,
- Vibration monitoring of main steam and feedwater lines,
- Steady-state data collection during incremental power ascension.

Individual Plant Examination

BWR plants use a probabilistic risk/safety assessment (PRA/PSA) to comply with the Individual Plant Evaluation (IPE) requirement. A plant-specific PRA/PSA is assessed for the effect of the CPPU. The assessment and any necessary updates are completed as required to support operation of the plant at a higher power level. The effect of CPPU on plant risk, including core damage frequency (CDF) and Large Early Release Fraction (LERF) are evaluated.

Operator Training and Human Factors

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

Plant Life

Various programs are implemented to monitor the aging of plant components, including Equipment Qualification, Flow Accelerated Corrosion, and Inservice Inspection. These programs are reviewed for the 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.

NRC and Industry Communications

The analysis, design, and implementation of CPPU is reviewed for compliance with the current plant licensing basis acceptance criteria. Compliance with existing regulatory requirements and operating experience in the nuclear industry is incorporated into this review process.

Emergency and Abnormal Operating Procedures

Emergency operating procedures (EOPs) include variables and limit curves, defining conditions where operator actions are indicated. Some of these variables and limit curves depend upon the value of rated reactor power. Changing some of the variables and limit curves will require modifying the values in the EOPs and updating supporting plant 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 the CPPU and updated, as necessary.

Abnormal operating procedures (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 the CPPU 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 included in Chapter 10. The licensing evaluations addressed in this section include:

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

General information is provided in this section to support utility licensing documentation required for the plant specific CPPU submittal. This general information is provided to assist the utility in the development of the environmental report, plant technical specification changes, and significant hazards assessment. The utility may elect to reference some or all of the information given in this section in the documentation supporting the plant specific licensing CPPU submittal.

11.1 EFFECT ON TECHNICAL SPECIFICATIONS

Implementation of CPPU requires revision of a number of the Technical Specifications. A list of Technical Specification changes will be included in the plant specific submittal.

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.

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 are examples of 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 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. A summary of the necessary plant modifications will be provided in the plant specific submittal.

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. The topics addressed are:

- Uprate Analysis Basis
- Margins
- Fuel Thermal Limits
- Makeup Water Sources

- Design Basis Accidents
- Challenges to Fuel
- Challenges to the Containment
- Design Basis Accident Radiological Consequences
- Anticipated Operational Occurrence Analyses
- Combined Effects
- Non-LOCA Radiological Release Accidents
- Equipment Qualification
- Balance-of-Plant
- Environmental Consequences
- Technical Specifications Changes

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 design basis accidents 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 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 guidelines of 10CFR100 Regulatory Guide 1.70 SAR Chapter 15 or plant specific acceptance 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.

12.0 REFERENCES

1. GE Nuclear Energy, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate", NEDC-32424P-A, February 1999.

2. 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.

3. GE Nuclear Energy, "General Electric Standard Application for Reactor Fuel", NEDE-24011-P-A and NEDE-24011-P-A-US, (latest approved revision).