ENCLOSURE 2

MFN 07-585

GEH Proprietary Markings for NRC Safety Evaluation of NEDC-33006P

Non-Proprietary Version

IMPORTANT NOTICE

This is a non-proprietary version of Enclosure 1 to MFN 07-585, from which the proprietary information has been removed. Portions of the enclosure that have been removed are indicated by an open and closed bracket as shown here [[ ]]

SAFETY EVALUATION BY

THE OFFICE OF NUCLEAR REACTOR REGULATION

LICENSING TOPICAL REPORT NEDC-33006P

"GENERAL ELECTRIC BOILING WATER REACTOR MAXIMUM EXTENDED LOAD LINE LIMIT ANALYSIS PLUS"

GENERAL ELECTRIC HITACHI NUCLEAR ENERGY AMERICA, LLC

Date: September 17, 2007

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EXECUTIVE SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) staff reviewed and approved the specific topics in the General Electric (GE) Hitachi Nuclear Energy America, LLC (GHNE, formerly known as GENE) licensing topical report (LTR) NEDC-33006P, "General Electric Boiling Water Reactor [BWR] Maximum Extended Load Line Limit Analysis [MELLLA] Plus," (MELLLA+) Revision 2 (Reference 1), dated November 2005, with the limitations and conditions as specified in Section 12.0 of the final SE. The LTR NEDC-33006P, Revision 2, proposed operation of GE-designed BWRs that implemented extended power uprates (EPUs) up to 20 percent above the original licensed thermal power (OLTP) at expanded power/flow operating domains. The expanded operating range is designed to enable plants that implemented EPU to operate at 120 percent of OLTP at flows ranging from 80 percent to 100 percent of rated core flow (CF).

BACKGROUND

BWRs were originally licensed to operate at rated power and CF (OLTP, 100 percent power/flow) along the flow control line. Currently, most BWRs are licensed to operate at the MELLLA operating domain, which is defined by an analytical line that passes through the 75 percent CF at the OLTP. Operation of BWRs at the MELLLA operating domain with increased maximum CF is referred to as the maximum extended operating domain (MEOD). The modified MEOD operating domain provides improved power ascension capability to full power and additional flow range at rated power.

Operation of BWRs requires that reactivity balance be maintained to accommodate fuel burn-up. BWR operators have typically two options to maintain this reactivity balance: (a) control rod movements or (b) flow adjustments. Because of the strong void reactivity feedback and its distributed effect through the core, flow adjustments are the preferred reactivity control method. Operation at low-flow conditions at rated power level also increases the fuel capacity factor through spectral shift and the increased flow region compensates for reactivity reduction due to fuel depletion during the operating cycle.

EPUs are implemented by extending the MELLLA operating domain up to EPU power levels. The extension of MELLLA line to EPU power levels reduces the available minimum CF window. In addition, the increased core pressure drop with EPU limits the recirculation flow capability. Thus, many EPU plants cannot achieve the increase CF operation. Consequently, EPU plants generally operate with minimum CF window (approximately 1 percent) and compensate for reactivity loss with control rod movement. Operation at the MELLLA+ expanded operating domain will provide a larger core window for EPU plants. Figure 1-1 shows the proposed MELLLA+ operating domain.

EPU OPERATING EXPERIENCE

Several BWRs are licensed for and have implemented EPU. Operating experience demonstrates that BWRs can operate at the EPU power levels with acceptable reactor core and fuel performance. EPU plants have experienced transients and the integrated systems and components actuated and performed as designed. The proposed MELLLA+ expanded operating domain will not change the EPU power levels at which plants operate. However, the operation at the higher power-to-flow ratio will affect the plant's core and fuel response and the associated fuel dependent analyses. Therefore, approval of MELLLA+ operation requires

demonstration that EPU plants can operate at the expanded power/flow domain and meet the fuel dependent regulatory and safety requirements.

MELLLA+ LTR SCOPE

LTR NEDC-33006P evaluates the impact of operation in the expanded operating domain on BWRs regarding: (1) safety systems and components capability and performance; and (2) response to the design bases and special events that demonstrate plants can meet the regulatory and safety requirements. The LTR dispositions the principle review topics generically or proposes that plant-specific analyses will be provided in the MELLLA+ applications to quantify the impact. This safety evaluation (SE) and evaluation of the RAI responses to the associated requests for additional information (see Appendices A, B, and C) provide the NRC staff assessment of the impact of operation at the MELLLA+ conditions on BWR performance and the capability of the plants to meet the safety and regulatory requirements.

SE REVIEW SCOPE

This SE evaluates the impact of operation at the expanded operating domain (MELLLA+) on the fuel dependent analyses and the associated safety systems and components. It also reviews and approves the plant-specific scope of fuel dependent safety analyses that will be submitted in the MELLLA+ safety analysis report (M+SAR). The principal topics covered in this SE are as follows:

1. Section 2.0, "Reactor Core and Fuel Performance," (1) evaluates the impact of operation at the higher power/flow fuel bundle conditions and rod line; and (2) proposes the analyses that will be provided to support the plant-specific MELLLA+ application. The principal topics covered are: Fuel Design and Operation (Section 2.1); Thermal Limit Assessment (Section 2.2); Reactivity Characteristics (Section 2.3); and Stability (Section 2.4). The fuel design and limiting thermal limits are performed on cycle-and core specific configuration during the standard reload process. The plant-specific applications will supplement the initial application and provide the cycle-specific fuel dependent analyses.

2. Section 3.0 "Reactor Coolant and Connected Systems," evaluates the impact of the operation at the expanded operating domain on the capability of the nuclear system pressure relief to meet its safety function and the plants American Society of Mechanical Engineers (ASME) overpressure (Section 3.1) response. The ASME overpressure analyses are performed on cycle and core configuration-specific bases during the standard reload. The plant-specific applications will supplement the initial MELLLA+ application and provide the plant-specific ASME overpressure response.

3. Section 4.0, "Engineered Safety Features," evaluates the capability of the emergency core cooling system (ECCS) capability (Section 4.1) and BWRs ECCS-loss-of-coolant accident (LOCA) response (Section 4.2) for operation at the MELLLA+ domain. The ECCS-LOCA analyses are not performed on cycle-specific basis. Therefore, the plant-specific applications will contain the ECCS-LOCA peak cladding temperature (PCT) results for operation at the expanded operating domain.

4. Section 5.0, "Instrumentation and Control," addresses adjustment and setpoint changes associated with the nuclear monitoring systems (Section 5.1) for operation at the expanded operating domain.

5. Section 6.0, "Electrical Power and Auxiliary Systems," topics include the capability and performance of the standby liquid control system (SLCS) (Section 6.5). The SLCS performance is associated with BWRs capability to meet the required redundant reactivity control system (cold shutdown margin (SDM) requirements) and the anticipated transient without scram (ATWS) mitigation requirements.

6. Section 9.0, "Reactor Safety Performance Evaluations," covers the anticipated operational occurrences (AOOs) (Section 9.1), and the special events such as the ATWS, and the ATWS with instability (Section 9.3). The plant-specific applications will supplement the initial application and provide the limiting AOO results for operation at the MELLLA+ boundary. The plant-specific submittal will also include ATWS analysis that will demonstrate that the plants can meet the ATWS acceptance criteria for operation at the expanded operating domains. The impact of MELLLA+ on BWR ATWS instability response and the effectiveness of the EPG ATWS/Stability mitigation actions will be demonstrated on a plant-specific basis.

SUMMARY OF IMPACT OF MELLLA+ ON FUEL DEPENDENT PLANT RESPONSE

MELLLA+ allows plants to operate at 120 percent of OLTP with CF as low as 80 percent of rated CF. The 120 percent rate power, 80 percent rated CF point corresponds to operation on approximately the 140 percent rod line. Many safety analyses are adversely impacted as a result of operation at the higher operating MELLLA+ domain. The safety analyses and plant response that are most severely impacted by the higher power-to-flow ratio allowed by MELLLA+ operation are mainly: thermal-hydraulic instability, ATWS, ATWS instability, and ECCS-LOCA. Some of the fuel dependent safety analyses that MELLLA+ operation significantly impacts are summarized below.

Impact on Stability Response

The regulation at Title 10 of the *Code of Federal Regulations* (10CFR) Part 50 Appendix A, General Design Criterion (GDC)-12 requires that oscillations are either not possible or can be reliably and readily detected and suppressed. Thermal-hydraulic instability analysis considers the two recirculation pump trip (2RPT) from the maximum allowable thermal power corresponding to the minimum allowable CF. By implementing MELLLA+, the allowable CF at rated power is further reduced, increasing the core two phase pressure drop, which decreases the stability margin. With operation at the higher 140 percent rod line, the reactor will settle at higher core power at the natural recirculation, following a 2RPT, as compared to MELLLA, leading to more unstable core conditions. Analytical evaluations of the impact of MELLLA+ operation on stability indicate that instabilities develop quickly, on the order of 10 seconds, following a 2RPT. Given the fast nature and rapid consequences of these transients under MELLLA+ conditions, the stability Long Term Solution (LTS) used on these plants must be approved for applicability to the reduced stability margin characteristic of MELLLA+ conditions. For RPTs initiating from the 55 percent MELLLA+ statepoint, the onset of instability will occur rapidly, limiting the effectiveness of operator actions to mitigate the plant instability response. Therefore, the NRC staff concluded that manual backup stability protection is not appropriate and a NRC-approved automatic backup stability protection must be implemented for MELLLA+ operation. The NRC staff approved GHNE's Detect and Suppress Solution – Confirmatory Density (DSS-CD) stability methodology presented in NEDC-33075P for application to MELLLA+ operation, including the availability of automatic back-up instability solution.

Impact on ATWS Response

MELLLA+ operation adversely impacts the plants ATWS analysis response, because of the operation at the higher rod line. ATWS-RPT initiated from the reduced CF at EPU power statepoint (e.g., 120 percent P/80 percent CF) is less effective relative to operation at the higher power/flow conditions as an initial condition. In addition, similar to stability discussion, due to operation at the 140 percent rod line, the reactor will settle at higher power levels but without scram. The short-term peak vessel overpressure becomes higher, as compared to MELLLA, due to the reduced power reduction capability afforded by the ATWS recirculation pump trip. Figure 9-1 schematically shows the reactor power reduction for RPT initiated from MELLLA relative to MELLLA+ minimum flow statepoint. In addition, Figure 9-11 shows the neutron flux response for MSIVC ATWS event followed by ATWS-RPT. As can be seen from Figure 9-3, the ATWS peak pressure response is higher for operation from MELLLA+ minimum CF statepoint relative to the MELLLA minimum CF statepoint at OLTP. The ATWS MSIVC data shown in Figure 9-3 and Figure 9-11 is based on ODYN calculations.

In addition, since the reactor is operating at a higher rod line, after RPT, the reactor core will settle at higher powers relative to operation at MELLLA+. The higher reactor power at natural recirculation increases the long-term heat load to the containment and results in higher suppression pool temperature. In its review, the NRC staff determined that for ATWS, the heat capacity temperature limit (HCTL) will be exceeded, at which point the operators are instructed to depressurize the reactor per plant EOPs. The HCTL value is plant-specific, and it depends roughly on the ratio of suppression pool volume to the reactor power. A typical value is in the range of 160°F.

However, the licensing ODYN code cannot model the depressurization or any ATWS water-level strategies, other than TAF+5. Therefore, the ODYN licensing calculation cannot model or simulate the actual plant conditions or operator actions as delineated by the EOP for ATWS. ODYN has been shown to be conservative for peak pressure relative to TRACG. The NRC staff concluded that the plant-specific applications will include ATWS sensitivity analyses simulating the ATWS scenario consistent with the plant-specific ATWS EOPs, including the water-level strategies employed at the plant, the depressurization if the HCTL is reached, and the associated operator actions and systems actuations

The predictions of consequences of the emergency depressurization are inconclusive. The sensitivity analyses show that the reactor can achieve hot shutdown conditions after the depressurization. However, both the NRC staff and GHNE calculations indicate there is a potential for re-criticality. Some of the TRACG simulations performed by GHNE indicate that, following the emergency depressurization, sufficient boron has been mixed into the core volume to maintain the reactor shutdown at the reduced pressure (approximately 100 pounds per square inch (psi)) by the combined effect of the mixed boron and the void fraction generated by decay heat. For these TRACG depressurization calculations, the containment limits are satisfied; fuel suffers dryout overheat due to core uncovery (see Figure 9-6), with a more severe transient for the lower-water-level control strategies, e.g., TAF-2 than for TAF+5 strategy. The NRC staff concludes that re-criticality after depressurization is not certain. It depends on plant- and event-specific parameters and/or assumed operator actions such as re-closing some SRVs after the pressure reaches the 50-psi target.

The plant-specific applications will include TRACG simulation following the EOPs, including depressurization, if the HCTL is reached.

The NRC staff approved the application of TRACG to ATWS instability in a generic review of the effectiveness of the mitigation actions (NRC staff SE dated February 5, 1994, approving GHNE LTRs NEDO-32047, "ATWS Rule Issues Relative to BWR Core Thermal-Hydraulic Stability,"

and NEDO-32164, "Mitigation of BWR Core Thermal-Hydraulic Instabilities in ATWS"). TRACG was also reviewed and approved for ATWS scenario up to the calculation of the peak pressures (LTR NEDE-32906P, "TRACG Application for Anticipated Operational Occurrences Transient Analysis," January 2000). In addition, in MFN 07-034 (Reference 27), GHNE committed to submit TRACG for ATWS application as the sole ATWS licensing evaluation code, and replace ODYN as the ATWS licensing code, with TRACG upon review and approval. In this review, the NRC staff had performed a limited evaluation of the boron mixing correlations modeled in TRACG. Since TRACG is a best estimate code that can model the ATWS scenario with more fidelity, including all the required operator actions and water level strategies, the NRC staff accepted the use of TRACG for performing the sensitivity analyses in addition to the currently licensed ODYN code. However, plant-specific applications will continue to use ODYN to demonstrate that the plants can meet the ATWS acceptance criteria, including the peak vessel pressure.

ATWS Suppression Pool Temperature

MELLLA+ implementation will adversely impact the ATWS response, including the suppression pool temperature. Simulation with and without depressurization predicts high final suppression pool temperatures of about 210 °F to 220 °F. Therefore, the NPSH is a concern for the required ECCS equipment under these conditions, including RHR. The high suppression pool temperatures will affect the operability and function of the safety system needed to mitigate the ATWS event. Plants may need containment overpressure credit in order to meet the NPSH requirements. In addition, for the safety systems such as HPCI, the temperature limits are set by the pump oil systems; thus, the containment overpressure credit may not alleviate the impact of the high suppression pool temperatures. In this case, the plants may rely on non-safety grade CST water supply, which is limited. The plant-specific applications will provide the impact of the MELLLA+ operation on the plant's capability to meet: (1) the suppression pool design limits; (2) the containment design limits; (3) the safety systems operability limits such as the NPSH requirements and equipment-specific temperature limits.

Impact on ATWS Instability Response

MELLLA+ operation adversely impacts BWR plants ATWS instability response due to the operation at the higher rod line (approximately 140 percent) and to some degree the core design associated with operation of EPU/MELLLA+ for 24-month cycle length.

The NRC staff review indicates that, in principle, MELLLA+ operation affects ATWS stability. Operation at the minimum CF at EPU power levels (120 percent OLTP, 80 percent CF) results in a significantly higher power following a 2RPT than when operating at MELLLA at OLTP or EPU. This higher power makes the final power even larger after the feedwater cool-down period; thus the unstable power oscillations are enhanced under MELLLA+, and their consequences would be expected to be more severe. However, TRACG simulations performed by GHNE demonstrated that the EPG mitigation actions are still effective in suppressing the oscillations during these ATWS events. Figure 9-19 shows the evolution of an ATWS-instability event without mitigation actions. The unstable power oscillations are allowed to grow to greater than 1000 percent. Following one of the power excursions, the fuel dries out and fails to re-wet. The resulting temperature excursion is sufficiently large to compromise the integrity of the fuel. The corresponding non-isolation ATWS analysis following the prescribed EPG mitigation actions (e.g., immediate water level reduction and boron injection) show that, for the particular reactor modeled, the power oscillations are adequately managed and the fuel integrity is not challenged. However, the NRC staff finds that the results of the ATWS-stability analysis have a large sensitivity to particular reactor conditions; therefore, a condition in the SE requires the

evaluation of the effectiveness of the ATWS-instability mitigation actions on a plant-specific basis.

Impact on ECCS-LOCA Response

The operation at reduced MELLLA+ CF conditions impact the large break ECCS-LOCA response. The reduced bundle power/flow condition causes early BT and affects the first PCT, which occurs during the flow coast down. Depending on the change in the first PCT and the plant-specific conditions, the second PCT could also change significantly. The changes in the design-basis accident (DBA) LOCA response for operation at the MELLLA+ reduced flow conditions are similar to the ECCS-LOCA response for the ELLLA and MELLLA reduced flow conditions.

The plant-specific applications will include DBA-LOCA analyses performed at the MELLLA+ minimum CF statepoint (120 percent OLTP, 80 percent CF) and the MELLLA+ knee statepoint (approximate OLTP, 55 percent CF). The most limiting DBA-LOCA PCT occurs at the 55 percent CF statepoint. However, for the DBA-LOCA calculations at the 55 percent CF statepoint, the analysis will take credit for the off-rated thermal limits multipliers. The multipliers will be applied at 55 percent or higher statepoints in the MELLLA+ domain. Taking credit of the off-rated limits requires that the core be operated with reduced bundle powers, such that lower thermal limits will not be exceeded. This approach differs from the current ECCS-LOCA assumptions, in which the rated thermal limits are conservatively applied to the reduced flow statepoints. Taking credit for the off-rated thermal limits will make the 55 percent CF statepoint DBA-LOCA response less bounding.

The higher EPU powers affect the small break LOCA, making the small break LOCA response more limiting for some plants. For operation at the MELLLA+ operating domain, the small break ECCS-LOCA response is not expected to change significantly from the EPU (120 percent OLTP, 99 percent CF) ECCS-LOCA response. However, for those plants in which the EPU small break LOCA is limiting or within [[ ]] of the limiting DBA PCT, small break LOCA analysis will be performed for the operation at the MELLLA+ domain. In addition, any limiting break location or single failure, which was previously shown to be within [[ ]] of the limiting case, will also be re-analyzed at the MELLLA+ low-flow conditions.

Impact of MELLLA+ on Reactor Core and Fuel Performance

With MELLLA+ operation, plants will be operating with maximum powered fuel bundles operating at high void conditions. With the operating window available, the spectral shift operation will result with the bundles operating with top-peaked power shapes, with upper part of the fuel bundles operating at high voids. Operation with high void conditions at the upper part of the fuel bundle, with top-peaked power shape will reduce the MCPR margins. In addition, MELLLA+ core thermal-hydraulic conditions have resulted in extension of the analytical methods outside the experience base and the applicability ranges. The NRC staff SE of LTR NEDC-33173P (Reference 38) provides assessment of the operation at high void conditions on the reactor core and fuel performance and extension of the analytical methods outside the validations ranges.

Conclusion of Impact of MELLLA+

The impact of operation at the MELLLA+ domain is covered in the applicable sections in this SE and the appended NRC staff evaluation of the responses to the requests for additional information (RAIs). The NRC staff review and approval of LTR NEDC-33006P, Revision 2, concluded with a number of limitations provided in Section 12. In addition, there are limitations that are applicable to MELLLA+ operation that are covered in separate LTRs as discussed below.

RELATED LTRS

There are several LTRs that cover specific review topics relevant to the approval of LTR NEDC-33006P. The limitations associated with these LTRs apply to the plant-specific MELLLA+ applications. The LTRs are as follows:

1. NEDC-33173P, "Applicability of GE Methods to Expanded Operating Domains," February 2006 (Reference 37). This LTR extends the use of GHNE's analytical methods and codes to MELLLA+. Plant-specific MELLLA+ applications must demonstrate compliance with the limitations in the NRC staff SE approving NEDC-33173P, or any supplements or revisions.

2. NEDC-33075P, "Detect and Suppress Solution-Confirmation Density Licensing Licensing Topical Report," Revision 5, July 2005 (Reference 45). This LTR presents stability detect and suppress methodology for application to MELLLA+ operation. The NRC staff reviewed and approved the stability methodology presented in this LTR for application to MELLLA+ operation (Reference 48). Specifically, the NEDC-33075P stability detect and suppress methodology ensures that the stability response for operation at the higher MELLLA+ rod line can be detected and suppressed such that GDC-12 requirements can be met. Stability detect and suppress methodology used to demonstrate the stability requirements can be met for operation in the MELLLA+ domain and is not limited to the DSS-CD, Revision 5, methodology. However, any detect and suppress methodology used must be specifically reviewed and approved for MELLLA+ operating conditions. The stability solution must also include a backup stability solution specifically reviewed and approved for MELLLA+ operation.

3. NEDE-33147, "DSS-CD TRACG Application," May 23, 2006 (Reference 47). GHNE used TRACG calculations to demonstrate that the DSS-CD stability solution can effectively detect and suppress instability events and meet the associated regulatory requirements. The NRC staff reviewed and accepted TRACG for this specific application (Reference 46).

Therefore, plant-specific MELLLA+ applications must comply with the limitations and conditions specified in the NRC staff SEs approving the latest versions of NEDC-33173P, NEDC-33075P, and NEDC-33147 (References 37, 45, and 47).

CONCURRENT CHANGES AND LICENSING PROCESS

The earlier versions of NEDC-33006P proposed a list of "separate effects" changes that could be implemented concurrently with the MELLLA+, but would be evaluated in a separate submittal. However, implementing all of these changes would have had a cumulative affect on the safety analyses that demonstrate the impact of MELLLA+ on the plant's response during steady-state, transients, accidents, and special events. Therefore, the plant-specific MELLLA+ application needs to demonstrate how the plant would be operated during the implementation of MELLLA+.

Section 1.2.1, "Concurrent Changes and Licensing Process," specifies the limitations related to proposed concurrent changes that affect the fuel dependent analyses or the safety system performance evaluations but are not considered in the MELLLA+ plant-specific response.

APPROVED VERSION

The NRC staff review and approval is based on Revision 2 of LTR NEDC-33006P (Reference 1). This revision incorporates changes culminating from the NRC staff review of the content of the earlier versions of the LTR.

ACKNOWLEDGEMENTS

The following NRC staff members are principal contributors to the review of NEDC-33006P:

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ACRONYMS

Term	Definition
ADS	Automatic Depressurization System
AL	Analytical Limit
AOO	Anticipated Operational Occurrence
AOP	Abnormal Operating Procedure
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
AV	Allowable Value
BOC	Beginning of Cycle
BT	Boiling Transition
BWR	Boiling Water Reactor
BWRVIP	Boiling Water Reactor Vessel and Internals Project
CF	Core Flow
CFR	Code Of Federal Regulations
CLTP	Current Licensed Thermal Power
CLTR	CPPU LTR, NEDC-33004P (Reference 5)
COLR	Core Operating Limits Report
CPPU	Constant Pressure Power Uprate
CPR	Critical Power Ratio
ΔCPR	Change in Critical Power Ratio
CS	Core Spray
CS/LPCS	Core Spray or Low Pressure Core Spray
DBA	Design-Basis Accident
DC	Direct Current
DIVOM	Delta CPR over Initial CPR vs. Oscillation Magnitude
DSS-CD	Detect And Suppress Solution–Confirmation Density
ECCS	Emergency Core Cooling System
ELLLA	Extended Load Line Limit Analysis
ELTR1	NEDC-32424P-A (Reference 3)
ELTR2	NEDC-32523P-A (Reference 4)
EMA	Equivalent Margins Analysis
EOC	End Of Cycle

Term	Definition
EOOS	Equipment Out-Of-Service
EOP	Emergency Operating Procedure
EPU	Extended Power Uprate
EPG	Emergency Procedure Guildline
ESF	Engineered Safety Features
FWCF	Feedwater Controller Failure
FWHOOS	Feedwater Heater(s) Out-Of-Service
FWT	Feedwater Temperature
GDC	Generic Design Criteria
GE	General Electric
GHNE	General Electric Nuclear Energy
GESTAR	GE Standard Application for Reactor Fuel
HCTL	Heat Capacity Temperature Limit
HPCI	High Pressure Coolant Injection
HPCS	High Pressure Core Spray
HSBW	Hot Shutdown Boron Weight
IASCC	Irradiation Assisted Stress Corrosion Cracking
IC	Isolation Condenser
ICA	Interim Corrective Actions
ICF	Increased Core Flow
IRM	Intermediate Range Monitor
LFWH	Loss of Feedwater Heater
LHGR	Linear Heat Generation Rate
LTR	Licensing LTR
LOCA	Loss-Of-Coolant Accident
LOOP	Loss Of Offsite Power
LPCI	Low Pressure Coolant Injection
LPCS	Low Pressure Core Spray
LPRM	Local Power Range Monitor
LRNBP	Generator Load Rejection Without Bypass
LSSS	Limiting Safety System Setting
LTS	Long Term Solution
MAPLHGR	Maximum Average Planar Linear Heat Generation Rate
MCPR	Minimum Critical Power Ratio
$MCPR_f$	Flow-dependent Minimum Critical Power Ratio
$MCPR_p$	Power-dependent Minimum Critical Power Ratio

Term	Definition
MELLLA	Maximum Extended Load Line Limit Analysis
MELLLA+	Maximum Extended Load Line Limit Analysis Plus
M+SAR	MELLLA+ Safety Analysis Report (Plant-specific Safety Analysis Report)
MEOD	Maximum Extended Operating Domain (MELLLA and ICF)
Mlbm	Millions Pound mass
MOC	Middle of Cycle
MSIV	Main Steam Isolation Valve
MSIVC	Main Steam Isolation Valve Closure
MSIVF	Main Steam Isolation Valve Closure With Scram On High Neutron Flux
MWt	Megawatt-Thermal
NFI	New Fuel Introduction
NMS	NMS
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
ODYN	GE methodology
OLMCPR	Operating Limit Minimum Critical Power Ratio
OLTP	Original Licensed Thermal Power
OPRM	Oscillation Power Range Monitor
PCT	Peak Cladding Temperature
PRA	Probabilistic Risk Assessment
PRFO	Pressure Regulator Failure Open
PUSAR	Power Uprate Safety Analysis Report
psi	Pounds Per Square Inch
psia	Pounds Per Square Inch - Absolute
psig	Pounds Per Square Inch - Gauge
P-T	Pressure-Temperature
RAI	Request for Additional Information
RBM	Rod Block Monitor
RCIC	Reactor Core Isolation Cooling
RCIS	Rod Control And Information System
RCPB	Reactor Coolant Pressure Boundary
RHR	Residual Heat Removal
RMS	Root Mean Square
RPS	Reactor Protection System
RPT	Recirculation Pump Trip

Term	Definition
RPV	Reactor Pressure Vessel
RRS	Reactor Recirculation System
RSLB	Recirculation Suction Line Break
RWE	Rod Withdrawal Error
RWM	Rod Worth Minimizer
SAFDL	Specified Acceptable Fuel Design Limit
SAG	Severe Accident Guideline
SAR	Safety Analysis Report
SBO	Station Blackout
SDC	Shutdown Cooling
SDM	Shut Down Margin
SE	Safety Evaluation
SL	Safety Limit
SLCS	Standby Liquid Control System
SLMCPR	Safety Limit Minimum Critical Power Ratio
SLO	Single (Recirculation) Loop Operation
SPC	Suppression Pool Cooling
SRLR	Supplemental Reload Licensing Report
SRM	Source Range Monitor
SRP	Standard Review Plan
SRV	Safety Relief Valve
SRVDL	Safety Relief Valve Discharge Line
T-M	Thermal – Mechanical
TAF	Top Of Active Fuel
TIP	Traversing In-Core Probe
TLO	Two (Recirculation) Loop Operation
TRACG	GE TRAC code
TS	Technical Specification
TSP	Trip Setpoint
TSV	Turbine Stop Valve
TTNBP	Turbine Trip without Bypass Failure
UFSAR	Updated Final Safety Analysis Report
UHS	Ultimate Heat Sink
USAR	Updated Safety Analysis Report
USE	Upper Shelf Energy

1.0 INTRODUCTION

By letter dated November 29, 2005, General Electric (GE) Hitachi Nuclear Energy America, LLC (GHNE, formerly known as GENE) submitted for U.S. Nuclear Regulatory Commission (NRC) staff review and approval licensing LTR (LTR) NEDC-33006P, Revision 2, "General Electric Boiling Water Reactor [BWR] Maximum Extended Load Line Limit Analysis [MELLLA] Plus [MELLLA+]," (Reference 1). Previous versions of this LTR were submitted to the NRC in 2002 and 2003 (References 2 through 4). This LTR defines the approach and provides the basis for an expansion of the core flow (CF) operating range for plants that have uprated power, either with or without a change in the operating pressure. This CF rate operating range expansion does not change the current plant vessel dome operating pressure. Supplemental information supporting the review of NEDC-33006P was provided to the NRC staff in References 5 through 32.

Power uprates in GE BWRs of up to 120 percent of original licensed thermal power (OLTP) have been based on the guidelines and approach provided in NEDC-32424P-A, February 1999 (ELTR1, Reference 33) and NEDC-32523P-A, February 2000, with Supplement 1, Volume I, February 1999, and Supplement 1, Volume II, April 1999 (ELTR2, Reference 34). The approach in ELTR1 and ELTR2 allows an increase in the maximum operating reactor pressure, when the reactor power is uprated. Subsequent to the approval of ELTR1 and ELTR2, GE developed an approach to uprate reactor power while maintaining the current reactor maximum operating reactor vessel dome pressure.

The current MELLLA operating range is characterized by the operating state point of reactor thermal power of 100 percent of OLTP at 75 percent of rated CF. Some plants currently combine the MELLLA operating region with increased CF resulting in an operating map called maximum extended operating domain (MEOD). Uprating to 120 percent OLTP using the MELLLA or MEOD boundary restricts the CF to 99 percent of rated CF at full power operation. This results in a reduced CF range that is available for flexible operation at the uprated power. LTR NEDC-33006P addresses the MELLLA plus (MELLLA+) operating improvement that provides an expansion of the operating boundary to permit operation of up to 120 percent OLTP with CF as low as 80 percent of rated CF.

LTR NEDC-33006P provides evaluations that demonstrate that the MELLLA+ operating range expansion can be accomplished within the applicable plant safety design criteria. Because the maximum thermal power and maximum CF rate do not change for MELLLA+, the effects are limited to the Nuclear Steam Supply System (NSSS), and primarily within the evaluation of core and reactor internals performance during postulated transient and accident events. In addition, many of the safety evaluations (SEs) and equipment assessments that have been previously performed for a power uprate are unaffected. This LTR dispositions these evaluations by generic assessments. Those evaluations that cannot be dispositioned by generic assessments will require the plant-specific evaluations to be documented in the plant-specific MELLLA+ safety analysis report (M+SAR). Licensees who reference this LTR in their request to implement MELLLA+ will document that the generic assessments of this LTR are applicable or provide a plant-specific evaluation.

1.1 BACKGROUND ON EXTENDED POWER UPRATE (EPU)

1.1.1 Generic EPU LTRs

The NRC staff reviewed and approved ELTR1 and ELTR2. The ELTRs, as supplemented, provide guidelines for plant-specific EPU applications involving dome pressure increase, and/or implementation of MELLLA operating domains and/or new fuel introduction (NFI). The ELTRs: (1) evaluated the impacts EPU operation would have on BWR response to the design basis and special events safety analyses, (2) provided generic bounding analyses; (3) evaluated the impact of EPU on equipment, components and systems important to safe operation of the plant, (4) identified the plant-specific supporting analyses that would be submitted in the EPU applications, and (5) presented the technical justifications supporting the specific principal topics found not to be significantly affected by the EPU operation. Therefore, the ELTRs provide the road map for the EPU applications, including resolution of any safety significant generic technical issues related to operation at EPU conditions.

Subsequently, the NRC staff reviewed and approved the constant pressure power uprate (CPPU) LTR (CLTR), NEDC-33004P-A, Revision 4 (Reference 35). The CLTR approval was limited to EPU applications that did not involve mixed-vendor transition cores and concurrent implementation of any changes in the operating conditions other than EPU. Since the CLTR is based on a limited set of analyses, any EPU application based on the CPPU cannot implement changes in the operating domains, introduce new fuel designs, or change the cycle length. Most EPU applications were based on ELTR1 and ELTR2.

1.1.2 Plant-Specific EPU Applications

Since the generic EPU LTRs established the required plant-specific assessment, the plant-specific EPU reviews focus on the results of the analyses for the specific application. The plant-specific reviews evaluate the plant's response to the design basis requirements and its capability to meet the acceptance criteria for each of the required safety analysis at the uprated conditions. The EPU application reviews also ensure that the key plant parameters (e.g., safety relief valve (SRV) tolerances, setpoint actuations, etc.) are consistent with the assumptions used in the plant-specific EPU analyses. Therefore, the plant-specific EPU reviews do not entail review of the methods used to perform the analyses but rather evaluate the plant-specific response to the operation at the uprated conditions.

1.1.3 NRC-Approved Analytical Methods and Codes

For BWR plants, the NRC staff reviews and approves the fuel vendor's licensing methodology, analytical methods, and codes used to perform the analyses supporting any licensing actions. GE's licensing methodology is specified in LTR NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (GESTAR II, Reference 36). Any changes to the licensing methodology, analytical methods, or codes require an amendment request. Amendment 22 to GESTAR II covers the required analyses for NFI and the analyses that are performed during the reload. The NRC staff reviews and approves LTRs that support the use of specific methods (e.g., critical power correlations, neutronic methods, stability solutions) or codes (e.g., TRACG for anticipated operational occurrence (AOO), TRACG for anticipated transient without scram (ATWS) peak pressure, ODYN, TASC). Upon approval, the LTRs and analytical methods are incorporated into the GESTAR II.

The SEs approving LTRs include limitations that delineate the conditions that warrant specific actions, such as obtaining measurement data or NRC-approval. The LTR, covering specific analytical methods or code systems, quantifies the accuracy of the methods or the code used and specifies the applicability ranges. Therefore, the use of NRC-approved analytical methods is contingent upon application of these methods and codes within the ranges for which the data was provided and against which the methods were evaluated. In those instances, the NRC staff SE does not contain specific limitations; the approval is based on the conditions and content of the plant-specific submittal.

1.1.4 Computer Codes and Methods

Section 1.3 of the power uprate safety analysis report (PUSAR) provides confirmation that the NRC-approved or industry-accepted computer codes and calculational techniques are used to demonstrate compliance with the applicable regulatory acceptance criteria. The PUSAR also states that the application of these codes to the EPU analyses complies with the limitations and conditions specified in the approving NRC SE where applicable for each code. Any exceptions to the use of the code or conditions of the applicable SE are noted in Table 1-1 of the PUSAR. In the plant-specific applications, Table 1-1 of the PUSAR lists the codes used to perform the safety analyses supporting the EPU operation. Therefore, in general, plant-specific licensing actions, including EPU reviews, do not entail review of the NRC-approved analytical methods and codes.

Section 1.1.3, "Computer Codes and Methods," of LTR NEDC-33006P proposes similar confirmation of methods applicability in the plant-specific MELLLA+ applications. The plant-specific MELLLA+ applications will reference the NRC-approved methods and codes used to perform the safety analyses and evaluations. The plant-specific M+SAR will provide a list of the computer codes used to perform the analyses and indicate the NRC-approval status.

During the review of LTR NEDC-33006P, the NRC staff discovered that for EPU and the proposed MELLLA+ operation, the NRC-approved analytical methods may: (1) be extended outside the applicability ranges; (2) not be adequately supported by the measurements qualification database; or (3) result in key parameters and assumptions being extended outside the acceptability ranges. As a result, GHNE submitted for review LTR NEDC-33173P, "Applicability of GE Methods to Expanded Operating Domains," (Reference 37) dated February 2006. In LTR NEDC-33173P, GHNE evaluated the impact of operation at higher void conditions characteristic of EPU and MELLLA+ operation on all of its licensing analytical methods. LTR NEDC-33173P was reviewed by the NRC staff and approval is pending with a number of limitations. These limitations are applicable to the present review of LTR NEDC-33006P.

In LTR NEDC-33006P, Revision 2, GHNE states that, "The Methods LTR NEDC-33173P (Reference [37]) documents all analyses supporting the conclusions in this section that the application ranges of GE codes and methods are adequate in the MELLLA+ operating domain. The range of mass fluxes and power/flow ratio in the GEXL database covers the intended MELLLA+ operating range. The database includes low flow, high qualities, and void fractions, although the void fraction is not measured in the test facility. Therefore, there are no restrictions on the application of the GEXL-PLUS correlation in the MELLLA+ operating domain."

Although GHNE does not measure the void fraction in the critical power ratio (CPR) experimental tests, the void fractions corresponding to the bundle test conditions can be calculated. The NRC staff also understands that the GEXL test database covers bundle flows lower than the bundle flows at the natural recirculation. For each bundle flow condition, the test

power level is increased, until boiling transition (BT) is reached, which would be expected to correspond to high void conditions. Using the test measurement data, the average void fractions at different axial elevation of the fuel bundle can be calculated to confirm that the within bundle void fraction ranges are covered by the GEXL-Plus database. In LTR NEDC-33173P, GHNE states: "the GEXL correlation database covers the EPU/MELLLA+ operating ranges."

GEXL-PLUS Limitation

The plant-specific application will confirm that for operation within the boundary defined by the MELLLA+ upper boundary and maximum CF range, the GEXL-PLUS experimental database covers the thermal-hydraulic conditions the fuel bundles will experience, including, bundle power, mass flux, void fraction, pressure, and subcooling. If the GEXL-PLUS experimental database does not cover the within bundle thermal-hydraulic conditions, during steady state, transient conditions, and DBA conditions, GHNE will inform the NRC at the time of submittal and obtain the necessary data for the submittal of the plant-specific MELLLA+ application.

In addition, the plant-specific application will confirm that the experimental pressure drop database for the pressure drop correlation covers the pressure drops anticipated in the MELLLA+ range.

With subsequent fuel designs, the plant-specific applications will confirm that the database supporting the CPR correlations covers the powers, flows and void fractions BWR bundles will experience for operation at and within the MELLLA+ domain, during steady state, transient, and DBA conditions. The plant-specific submittal will also confirm that the NRC staff reviewed and approved the associated CPR correlation if the changes in the correlation are outside the GESTAR II (Amendment 22) process. Similarly, the plant-specific application will confirm that the experimental pressure drop database does cover the range of pressures the fuel bundles will experience for operation within the MELLLA+ domain.

1.1.5 Related LTRs

There are several LTRs that cover specific review topics relevant to the approval of NEDC-33006P. The limitations associated with these LTRs apply to the plant-specific MELLLA+ applications. The LTRs are as follows:

1. NEDC-33173P, "Applicability of GE Methods to Expanded Operating Domains," February 2006 (Reference 37). This LTR extends the use of GHNE's analytical methods and codes to MELLLA+. Plant-specific MELLLA+ applications must demonstrate compliance with the limitations in the NRC staff SE approving NEDC-33173P, or any supplements or revisions.

2. NEDC-33075P, "Detect and Suppress Solution-Confirmation Density Licensing Licensing Topical Report," Revision 5, July 2005 (Reference 45). This LTR presents stability detect and suppress methodology for application to MELLLA+ operation. The NRC staff reviewed and approved the stability methodology presented in this LTR for application to MELLLA+ operation. Specifically, the NEDC-33075P stability detect and suppress methodology ensures that the stability response for operation at the higher MELLLA+ rod line can be detected and suppressed such that GDC-12 requirements can be met. Stability detect and suppress methodology used to demonstrate the stability requirements can be met for operation in the MELLLA+ domain and is not limited to the DSS-CD, Revision 5, methodology. However, any detect and suppress methodology used must be specifically reviewed and approved for MELLLA+ operating conditions. The stability solution must also include a backup stability solution specifically reviewed and approved for MELLLA+ operation.

3. NEDE-33147, "DSS-CD TRACG Application," May 23, 2006 (Reference 47). GHNE used TRACG calculations to demonstrate that the DSS-CD stability solution can effectively detect and suppress instability events and meet the associated regulatory requirements. The NRC staff reviewed and accepted TRACG for this specific application.

Related LTRs Limitation

Plant-specific MELLLA+ applications must comply with the limitations and conditions specified in and be consistent with the purpose and content covered in the NRC staff SEs approving the latest version of the following LTRs: NEDC-33173P, NEDC-33075P, and NEDC-33147 (References 37, 45, and 47).

Table 1-1 Computer codes used for CPPU

Task	Computer Code	Version or Revision	NRC Approved	Comments
Nominal Reactor Heat Balance	ISCOR	09	Y(2)	NEDE-24011P Rev. 0 SER
Reactor Core and Fuel Performance	TGBLA PANACEA GESAM	06 11 01	Y Y (5) Y(3)	NEDE-30130-P-A NEDE-30130-P-A NEDO-10958-A
Reactor Power/Flow Map	BILBO	04V	NA	(1); NEDE-23504, February 1977
Thermal Hydraulic Stability	ODYSY	05	Y	NEDC-32992P-A
Reactor Vessel Fluence	DORTG01 TGBLA	1 6	N Y	(14) (15)
Reactor Internal Pressure Differences	ISCOR LAMB TRACG	09 07 02	Y(2) (4) Y	NEDE-24011P Rev. 0 SER NEDE-20566-P-A NEDE-32176P, Rev 2, Dec 1999 NEDC-32177P, Rev 2, Jan 2000 NRC TAC No M90270, Sep 1994 (13)
Containment System Response	SHEX M3CPT LAMB	05 05 08	Y Y (4)	(8) NUREG-0661 NEDE-20566-P-A
Transient Analysis	PANACEA ISCOR ODYN SAFER TASC	11 09 10 04 03A	Y Y(2) Y Y(6) Y	NEDE-30130-P-A (5) NEDE-24011P Rev. 0 SER NEDO-24154-A NEDC-32424P-A, NEDC-32523P-A, (9), (10), (11) NEDC-32084P-A, Rev. 2, July 2002
Anticipated Transient Without Scram	ODYN STEMP PANACEA ISCOR TASC SHEX	10 04 11 9 03A 05	Y (7) Y Y (2) Y Y	NEDE-24154P-A Supp. 1, Vol. 4 NEDE-30130-P-A NEDE-24011P Rev. 0 SER NEDC-32084P-A, Rev. 2, July 2002 (8)
Station Blackout	SHEX	05	Y	(8)
Appendix R Fire Protection	GESTR SAFER SHEX	08 04 05	(6) (6) (8)	NEDE-23785-1-PA, Rev. 1 (9) (10) (11)
Reactor Recirculation System	BILBO	04V	NA	(1) NEDE-23504, February 1977
ECCS-LOCA	LAMB GESTR SAFER ISCOR TASC	08 08 04 09 03A	Y Y Y Y(2) Y	NEDO-20566A NEDE-23785-1-PA, Rev. 1 (9) (10) (11) NEDE-24011P Rev. 0 SER NEDC-32084P-A, Rev. 2, July 2002
Fission Product Inventory	ORIGEN2	2.1	N	Isotope Generation and Depletion Code
High Energy Line Break	COMPARE-MOD 1	A	N(12)	LA-7199-MS
Probabilistic Risk Assessment	MAAP	4.0.4	N	(16)

1.1.6 MELLLA+ Operating Domain Overview

Figure 1-1 of this SE shows a power-flow map for a typical BWR. BWRs operating at the OLTP typically operate in a power-flow range identified as the MELLLA range, which is characterized by the operating statepoint of reactor thermal power of 100 percent of OLTP at 75 percent flow (point C of Figure 1-1). Some plants currently combine the MELLLA operating region with increased core flow (ICF) resulting in an operating map called MEOD, which increases the power-flow range up to 107 percent flow (point A of Figure 1-1).

Power uprates in BWRs of up to 120 percent of OLTP have been reviewed and approved. Some early implementations used the approach described in LTR NEDC-32424P-A (Reference 33) and LTR NEDC-32523P-A (Reference 34). Most recent power uprates maintain a constant dome pressure, as described in NEDC-33004P-A (Reference 35). These uprates all consist of an extension of the MELLLA or MEOD boundary along the flow control line, so that a flow reduction or a recirculation pump trip (RPT) would revert approximately to the pre-OLTP operation statepoints. As seen in Figure 1-1, uprating to 120 percent OLTP using the MELLLA or MEOD boundary restricts the CF to be greater than 99 percent of rated full power operation (point B of Figure 1-1), which results in a reduced CF range available for flexible operation at the uprated power.

Day-to-day operation of nuclear reactors requires that reactivity balance be maintained to accommodate fuel burn-up. The BWR operators have typically two options to maintain this reactivity balance: (a) control rod movements or (b) flow adjustments. Because of the strong void reactivity feedback and its distributed effect all over the core, flow adjustments are the preferred reactivity control method. Control rod movements are typically performed a few times during the cycle to accomplish larger reactivity changes and the desired burn-up profiles. Because of the strong local power changes that may result from control rod motion and its local effect on the fuel, control rod movements should be performed very slowly and at a reduced power level; otherwise, fuel clad failures may occur.

The preferred reactivity control method, which has been used for many years in BWRs, is to set up a target control rod pattern at a low power level, increase the power to full licensed conditions and control reactivity by increasing flow over a period of several months. When the burn-up reactivity can no longer be adjusted using flow, the power level is reduced, the next target control rod sequence is achieved, the power is increased back to the licensed level, and flow control continues to maintain power. Figure 1-2 of this SE illustrates how a reactor operator can use the flow-control window to adjust changes in reactivity caused by burn-up.

As seen in Figure 1-1, the flow control window in EPU reactors is very small (approximately 1 percent flow). Therefore, reactor operators are forced to either move control rods very often or allow power changes as burn-up takes place. In a typical EPU reactor, the control rods must be repositioned almost on a weekly basis to maintain power at the licensed level.

MELLLA+ attempts to address this flow control issue by increasing the operating range to point D of Figure 1-1 (120 percent OLTP, 80 percent flow); thus creating a 20 percent flow-control window. The EPU plants operating in the MELLLA+ range will require significantly lower number of control rod movements than presently licensed EPU plants. This represents a significant improvement on operating flexibility. It also provides safer operation, because reducing the number of control rod manipulations: (a) minimizes the likelihood of fuel failures and (b) reduces the likelihood of accidents initiated by reactor maneuvers required to achieve an operating condition where control rods can be extracted. However, this safety increase must be compared to other factors such as the higher power effect on transients and the fact that more channels are placed closer to limits by the power-profile flattening.

A secondary benefit from MELLLA+ operation is spectral shifting. Operation at high power-to-flow ratios results in high void fractions, and the reduced water-moderation of neutrons increases the neutron average energy. At higher neutron energies, the Uranium (U)-238 absorption cross-section increases, and more Plutonium (Pu)-239 is produced. Since Pu-239 is a fissile isotope, it increases the core reactivity and, essentially, adds production days to the fuel cycle. Towards the end of cycle, approximately 30 percent of the nuclear energy is produced by fission of the Pu-239 as opposed to U-235. Thus, operation at the increased power-to-flow ratio allowed by MELLLA+ provides a significant economic advantage.

1.2 LICENSING APPROACH

LTR NEDC-33006P describes the generic guidelines, evaluations, criteria, process, and scope of work that would be needed to support operation in the MELLLA+ operating domain. The LTR addresses the safety aspects of the plant that are affected by operation at this increased power and reduced flow, including the NSSS and balance-of-plant (BOP) systems. The LTR defines the methodology, analysis assumptions, and acceptance criteria to be used in plant-specific M+SAR.

LTR NEDC-33006P provides the proposed format for the plant-specific M+SAR. The proposed format, scope, and content of the LTR are similar to previous GE generic power uprate LTRs (e.g., NEDC-33004P-A (Reference 35)). The plant-specific M+SAR will follow the same scope, content, and structure as LTR NEDC-33006P, as supplemented by the conclusions of this SE.

The applicable sections of this SE cover the fuel-dependent analysis. This SE delineates the bases of the approval and the scope and content of M+SAR. The "-A" version of NEDC-33006P will revise the LTR NEDC-33006P and ensure that there are no inconsistencies between the content and scope proposed in LTR NEDC-33006P and the bases of the NRC staff's SE approving LTR NEDC-33006P.

1.2.1 Concurrent Changes and Licensing Process

The earlier versions of LTR NEDC-33006P proposed a list of "separate effects" changes that could be implemented concurrently with the MELLLA+, but would be evaluated in a separate submittal. However, implementing all of these changes would have had a cumulative affect on the safety analyses that demonstrate the impact of MELLLA+ on the plant=s response during steady-state, transients, accidents, and special events. Therefore, the plant-specific MELLLA+ application needs to demonstrate how the plant would be operated during the implementation of MELLLA+.

Concurrent Changes Limitation

a) The plant-specific analyses supporting MELLLA+ operation will include all operating condition changes that are implemented at the plant at the time of MELLLA+ implementation. Operating condition changes include, but are not limited to, those changes that affect, an increase in the dome pressure, maximum CF, fuel cycle length, or any changes in the licensed operational enhancements. For example, with an increase in dome pressure, the following analyses must be analyzed: the ATWS analysis, the American Society of Mechanical Engineers (ASME) overpressure analyses, the transient analyses, and the emergency core cooling system-loss-of-coolant accident (ECCS-LOCA) analysis. Any changes to the safety system settings or any actuation setpoint changes necessary to operate with the increased dome pressure must be included in the evaluations (e.g., safety relief valve (SRV) setpoints).

b) For all topics in LTR NEDC-33006P that are reduced in scope or generically dispositioned, the plant-specific application will provide justification that the reduced scope or generic disposition is applicable to the plant. If changes that invalidate the LTR dispositions are to be implemented at the time of MELLLA+ implementation, the plant-specific application will provide analyses and evaluations that demonstrate the cumulative effect with MELLLA+ operation. For example, if the dome pressure is increased, the ECCS performance will be evaluated on a plant-specific basis.

c) Any generic bounding sensitivity analyses provided in LTR NEDC-33006P will be evaluated to ensure that the key plant-specific input parameters and assumptions are applicable and bounded. If these generic sensitivity analyses are not applicable or additional operating condition changes affect the generic sensitivity analyses, a plant-specific evaluation will be provided. For example, with an increase in the dome pressure, the ATWS sensitivity analyses that model operator actions (e.g., depressurization if the heat capacity temperature limit (HCTL) is reached) needs to be reanalyzed, using the bounding dome pressure condition.

d) If a new GE fuel product line or another vendor's fuel is loaded at the plant, the applicability of any generic sensitivity analyses supporting the MELLLA+ application shall be justified in the plant-specific application. If the generic sensitivity analyses cannot be demonstrated to be applicable, the analyses will be performed including the new fuel. For example, the ATWS instability analyses supporting the MELLLA+ condition are based on the GE14 fuel response. New analyses that demonstrate the ATWS instability performance of the new GE fuel or another vendor's fuel for MELLLA+ operation shall be provided to support the plant-specific application.

e) If a new GE fuel product line or another vendor's fuel is loaded at the plant prior to a MELLLA+ application, the analyses supporting the plant-specific MELLLA+ application will be based on a specific core configuration or bounding core conditions. Any topics that are generically dispositioned or reduced in scope in LTR NEDC-33006P will be demonstrated to be applicable, or new analyses based on the specific core configuration or bounding core conditions will be provided.

f) If a new GE fuel product line or another vendor's fuel is loaded at the plant prior to a MELLLA+ application, the plant-specific application will reference an NRC-approved stability method supporting MELLLA+ operation, or provide sufficient plant-specific information to allow the NRC staff to review and approve the stability method supporting MELLLA+ operation. The plant-specific application will demonstrate that the analyses and evaluations supporting the stability method are applicable to the fuel loaded in the core.

g) For MELLLA+ operation, core instability is possible in the event a transient or plant maneuver places the reactor at a high power/low-flow condition. Therefore, plants operating at MELLLA+ conditions must have a NRC-approved instability protection method. In the event the instability protection method is inoperable, the applicant must employ an NRC-approved backup instability method. The licensee will provide technical specification (TS) changes that specify the instability method operability requirements for MELLLA+ operation, including any backup stability protection methods.

1.2.2 MELLLA+ LTR Approach

LTR NEDC-33006P assesses the BWR plant safety, system, and component performance, identifies the principal topics of review that are affected by MELLLA+, and those that are not significantly affected. For specific areas of the BWR safety design, the LTR provides generic bounding evaluations. For those evaluations that are not categorized as generically dispositioned, a plant-specific evaluation will be required and will be documented in the plant-specific M+SAR submittal consistent with the contents, structure, and level of detail indicated in the MELLLA+ SE.

1.2.2.1 Generic Assessment Approach

The LTR NEDC-33006P generically dispositions certain topics by:

1. Providing or referencing a bounding analysis for the limiting conditions,
2. Demonstrating that there is a negligible effect due to MELLLA+,
3. Identifying the portions of the plant that are unaffected by the MELLLA+ power-flow map operating range expansion, or
4. Demonstrating that the sensitivity to MELLLA+ is small enough that the required plant cycle specific reload analysis process is sufficient and appropriate for establishing the MELLLA+ licensing basis (as defined in GESTAR II, Reference 36).

LTR NEDC-33006P provides a phenomenological discussion of the effect of MELLLA+ on the evaluation results. These sections reference the applicable experience base and associated supporting information. The M+SAR will confirm and document the applicability of the generic assessments.

The LTR NEDC-33006P generic dispositions are based on the bounding analysis, negligible effect, unaffected, and reload dependent assessments discuss below.

1.2.2.2 Bounding Analysis

Those safety analyses and evaluations of equipment, component, and systems performance that are generically dispositioned will not be included in the plant-specific M+SAR, because the:

1. Uprate assessments in CLTR, ELTR1, or ELTR2 are bounding,
2. Specific MELLLA+ generic studies are provided in MELLLA+ LTR, or
3. Previous studies in generic or plant-specific safety analysis report submittals are shown to be applicable.

1.2.2.3 Negligible Effect

For those safety analyses and evaluations of equipment, component and systems performance that are negligibly affected by MELLLA+ operation, the specific supporting evaluation will be provided. The applicable sections discuss the bases for the negligible assessment and provide the supporting, current experience and/or analyses. Where applicable, LTR NEDC-33006P references the CLTR, ELTR1, or ELTR2 evaluations that support the conclusion of negligible effect. Any plant system design that falls outside of the current basis for a "negligible effect" will be addressed in the plant-specific submittal.

1.2.2.3.1 Unaffected

The LTR NEDC-33006P notes that MELLLA+ operation directly affects the core and some aspects of the NSSS and it does not change the thermal power, normal operating pressure, steam flow, feedwater flow, or feedwater temperature (FWT). The Power Conversion Systems, Section 7.0, and Electrical Power and Auxiliary Systems, Section 6.0 of the LTR, are examples of subjects where there is no change resulting from the MELLLA+ operating range expansion.

1.2.2.3.2 Reload Dependent

LTR NEDC-33006P proposed to disposition the fuel-dependent analyses to the Standard Reload Process, stating that:

> The reload dependent evaluation process requires that the reload fuel design, core loading pattern, and operational plan be established so that analyses can be performed to establish operating limits for the cycle-specific core configuration. The reload analysis process is required to demonstrate that the core design, including the operating limits in the MELLLA+ operating range, will meet all of the applicable NRC evaluation criteria and limits documented in Reference 4. [[

]] The MELLLA+ operating range expansion cannot be implemented unless the appropriate reload core analysis is performed, the core and fuel operating limits are appropriately established, and the criteria and limits in Reference 4 are satisfied. Based upon 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).

The NRC staff agrees that the cycle- and core-specific analysis will not be available in the initial submittal of the plant-specific M+SAR. In addition, in accordance with the NRC-approved licensing process specified in GESTAR II, the reload fuel dependent analyses will be performed and documented in the supplemental reload licensing report (SRLR). However, the NRC staff notes that the SRLR is not submitted unless the NRC staff specifically requests it in request for additional information (RAI). For the CPPU applications, the core and fuel performance assessments are performed on a representative core and the actual core and fuel performance assessments deferred to the reload. Therefore, MELLLA+ LTR proposes that the NRC staff approve an MELLLA+ application without reviewing the plant's response for two significant operational changes. The NRC staff finds that the proposed disposition of the fuel- and cycle-dependent analyses to the standard reload process would not meet the agency's safety goals.

Moreover, the generic disposition to the reload rejected by the staff was based on the assessment that the plant's core and fuel response to the MELLLA+ conditions would not be significantly different from responses during EPU operation. However, the NRC staff finds that the high bundle power/flow conditions for MELLLA+ operation will reduce the margins to the thermal limits. The NRC staff RAI 24 discusses the disposition of the fuel- and cycle-dependent analyses to the reload. In response to NRC staff RAI 24 (Reference 30), GHNE accepted that the plant-specific MELLLA+ applications will provide the thermal limits assessment and the transient analysis results.

Reload Analysis Submittal Limitation

The plant-specific MELLLA+ application shall provide the plant-specific thermal limits assessment and transient analysis results. Considering the timing requirements to support the reload, the fuel- and cycle-dependent analyses including the plant-specific thermal limits

assessment may be submitted by supplementing the initial M+SAR. Additionally, the SRLR for the initial MELLLA+ implementation cycle shall be submitted for NRC staff confirmation.

1.2.2.3.3 Thermal Limits and Transient Results Limitation

As described in the GHNE response to RAI 24 (Reference 30), the plant-specific MELLLA+ application will provide the plant-specific thermal limits assessment and transient analysis results. The fuel- and cycle-dependent analyses including the plant-specific thermal limits assessment can submitted by supplementing the initial M+SAR.

The LTR NEDC-33006P states that if the generic assessment is fuel design dependent, this assessment is applicable only to GE/Global Nuclear Fuel (GNF) fuel designs through GE14, analyzed with GE methodology. It adds that the effect of MELLLA+ on future GE/GNF fuel designs will be addressed during the assessment of the new fuel design consistent with the requirements of GESTAR II (Reference 36). This statement is unclear as to what analyses will be provided in the M+SAR if the core is loaded with new GE fuel designs. For clarity, the NRC staff reiterates that the scope of fuel-dependent analyses and the results will not be deferred to the NFI process but will be provided in the M+SAR. The fuel-dependent analyses that are also cycle-dependent and performed during the standard reload analysis can be submitted after the SRLR is available.

1.2.2.3.4 Plant-Specific Evaluation

All topics that are not categorized as generic will require a plant-specific evaluation and will be documented in the plant-specific M+SAR submittal. The LTR NEDC-33006P provides an assessment of the expected MELLLA+ effect on the plant and also provides guidelines as to the plant-specific evaluations that will be provided in the M+SAR.

1.2.2.4 Cores Loaded with Non-GE Fuel

LTR NEDC-33006P states that if another vendor's fuel design is considered as part of the MELLLA+ operating range expansion, fuel design dependent assessments must be separately evaluated and justified on a plant- and fuel-specific basis.

It is important to note that the LTR NEDC-33006P fuel-dependent assessments are limited to the GE14 fuel designs. Therefore, this SE did not cover mixed vendor cores or cores consisting exclusively of another vendor's fuel, because no generic assessments were provided in LTR NEDC-33006P that demonstrates the plants' fuel-dependent response. Therefore, plants loaded with non-GE fuel or using another vendor's analytical methods and codes will provide all the fuel dependent analyses to demonstrate the safe operation of the plants at MELLLA+ conditions. This includes the fuel and core performance, the thermal margins assessments, the off-rated limit analyses, the thermal mechanical overpower, the overpressure, the LOCA analysis (full break spectrum), the ATWS, ATWS instability, and the stability responses. The analyses assumptions and calculational methods will be consistent with the content and scope of LTR NEDC-33006P and this SE. The plant-specific application will account for the topics covered in the NRC staff RAIs. Some of the key calculational methodology and assumptions important to the MELLLA+ operation include but are not limited to:

1. performing the analyses (e.g., LOCA, SLMCPR, transients) at the limiting MELLLA+ statepoints;

2. performing the LOCA analyses for top peaked power shape;

3. performing thermal and mechanical overpower analysis; and

4. demonstrating that the ATWS acceptance criteria can be met, including the core depressurization if the suppression pool temperature reaches the HCTL. The HCTL is defined so that transferring all the stored energy of the pressurized primary system to the suppression pool will not result in containment integrity violation.

1.2.2.5 NEDC-33173 Applicability

LTR NEDC-33173P covers the applicability of the analytical methods and codes used to perform the safety analyses for MELLLA+ operations and is limited to GE analytical methods and codes. Thus, the plant-specific M+SAR will either demonstrate compliance with the applicable limitations in the SE approving LTR NEDC-33173P in addition to the limitations in this SE or establish that the limitation is not applicable.

1.3 OPERATING CONDITIONS AND CONSTRAINTS

1.3.1 Power/Flow Map

Figure 1-1 shows the MELLLA+ operating domain. The MELLLA+ operating domain is bounded by an analytical line that extends from 55 percent CF to the minimum CF statepoint (e.g., 80 percent CF) at EPU power level and the CF window at EPU power level (80 percent to rated or ICF at EPU power level).

Most BWRs implemented the MELLLA operating domain. The LTR NEDC-33006P presents the MELLLA and MELLLA+ analytical lines. The MELLLA upper boundary core power, P (percent rated), as a function of CF, W_T (percent rated), is defined as:

[[

]] Although the load line is influenced by plant-specific operating factors such as the FWT and the core size, changes in the load line due to core characteristics (e.g., reactivity coefficients and power distribution) can be represented using the [[

]]

The MELLLA+ region extends down to 55 percent CF. The MELLLA+ was not extended below 55 percent CF due to stability considerations. Plant power/flow maneuvers near the upper boundary of MELLLA+ near full power are intended to be performed above 55 percent CF statepoint. If the reactor operating conditions following an unplanned event stabilize at a power/flow point outside the allowed operating domain, the operators must maneuver the plant back into the analyzed and licensed domain. This is consistent with the current plant procedures and operation. However, in the initial implementation of MELLLA+, the approving NRC staff SE will flag as an inspection item the operator training modules and awareness of a potential operation outside the licensed MELLLA+ domain.

1.3.2 Core and Reactor Conditions

Table 1-2 of this SE presents the plant parameters for a BWR/6 plant operating at the: (1) 120 percent of OLTP, 99 percent CF statepoint; (2) 120 percent OLTP, 80 percent CF statepoint; and (3) 97 percent of the OLTP, 55 percent CF statepoint. The LTR NEDC-33006P states that the differences shown in Table 1-2 represent the characteristics of other BWR plants, however the core operating conditions represent the maximum allowed power-to-flow ratio. For operation in the MELLLA+ minimum flow statepoint, the changes in the reactor heat balance are primarily due to the decrease in the recirculation flow. As seen in Table 1-2 below, the parameter for 99 percent CF and 80 percent CF decreases from 83.7 percent to 67.6 percent Mlb/Hr, while the feedwater temperature remains at 430 EF, which decreases the core inlet enthalpy from 525.2 Btu/Lb to 519 Btu/Lb. The core average exit void fraction changes from 73 percent to 77 percent. However, it is important to note that the exit void fraction of the maximum powered bundles would be above 90 percent, depending on the plant-specific core configuration.

Table 1-2 Comparison of Thermal-Hydraulic Parameters

Parameter	MELLLA 120 percent OLTP, 99 percent CF Normal FWT	MELLLA 120 percent OLTP, 99 percent CF Reduced FWT	MELLLA+ 120 percent OLTP, 80 percent CF Normal FWT	MELLLA+ 97 percent OLTP, 55 percent CF Normal FWT
Thermal Power (MWt)	3473	3473	3473	2807
Steam Flow rate (Mlb/Hr)	15.15	14.18	15.15	11.83
Dome Pressure (psia)	1040	1040	1040	1004
FWT (°F)	430	380	430	406
CF (Mlb/Hr)	83.7	83.7	67.6	46.5
Core Inlet Enthalpy (Btu/Lb)	525.2	517.8	519.0	504.1
Core Pressure Drop (psi)	26.1	25.4	19.3	11.2
Core Average Void Fraction	0.52	0.49	0.55	0.56
Average Core Exit Void Fraction	0.73	0.71	0.77	0.78

Table 1-3 of this SE shows a comparison of void fraction levels calculated at MELLLA+, MELLLA, and OLTP (100 percent power/100 percent flow) at different core locations, including the bypass region and the hot channel. From the data in these two tables, the NRC staff concludes that MELLLA+ operation increases the void fraction significantly (up to 93 percent voids for the hot channel) when compared to MELLLA and OLTP operation. This void fraction increase was one of the factors that triggered the methods review documented in LTR NEDC-33173P (Reference 37) and the NRC staff SE approving LTR NEDC-33173 (Reference 38).

Table 1-3 Bypass Void Fractions Calculated for Different Reactor Operating Domains

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The NRC staff concurs with GE's conclusion that decay heat is principally a function of the reactor power level and irradiation time. MELLLA+ does not alter either of these two parameters, and therefore there is no first order affect on decay heat. Additional parameters that have a second order impact on decay heat include: enrichment, exposure, void fraction, power history, cycle length, and refueling batch fraction.

1.3.3 Operational Enhancements

Table 1-4 below shows the operational flexibility not allowed in MELLLA+ operation. The BWR plants are allowed to operate with equipment out-of-service (EOOS), provided the safety analysis supporting the operation with the equipment configuration demonstrates that the

fuel-dependent regulatory and safety requirements can be met. The cycle- and core-specific reload analyses are performed assuming the EOOS.

Table 1-4 Excluded Options in the MELLLA+ Domain

Operational Enhancements Not Allowed in MELLLA+ Operating Region
Feedwater Heater Out-of-Service (FWHOOS)
Single Loop Operation (SLO)

[[

]] The inlet subcooling for the 55 percent CF is lower than the EPU statepoint subcooling, but prohibiting FWHOOS ensures the initial subcooling does not decrease further, degrading the stability response in the event of recirculation pump trip (RPT).

LTR NEDC-33006P states that single loop operation (SLO) in the MELLLA+ region is not proposed; however the available operating range for SLO in the MELLLA+ region may be considered on plant-specific basis. The CF attainable with the SLO is typically 50 percent of rated CF and would not be expected to be higher than 60 percent of rated flow. Therefore, for some BWR plants, SLO flow range could place the plant outside the proposed MELLLA+ operating domain. Since the MELLLA+ line is at a higher rod line, a 2RPT will settle the reactor at higher power/flow conditions, adversely affecting the stability response. The 2RPT is potentially higher from SLO configuration. In addition, the higher flow noise level and the reverse flow, during SLO operation, can potentially affect the accuracy of the CF measurement, which could impact establishing the core operating statepoint. Therefore, the NRC staff finds that SLO operation is not prudent until sufficient experience is gained in the operation at the new MELLLA+ domain.

The M+SAR will identify the applicable plant-specific operational flexibilities allowed for operation at the MELLLA+ domain. The acceptability of any proposed SLO operation will be evaluated on plant-specific bases.

The following limitations apply to the operational flexibilities that are prohibited in the MELLLA+ operation:

Operating Flexibility Limitations:

a) The licensee will amend the TS LCO for any equipment out-of-service (i.e., SLO) or operating flexibilities prohibited in the plant-specific MELLLA+ application.

b) For an operating flexibility, such as FWHOOS, that is prohibited in the MELLLA+ plant-specific application but is not included in the TS LCO, the licensee will propose and implement a license condition.

c) The power flow map is not specified in the TS; however, it is an important licensed operating domain. Licensees may elect to be licensed and operate the plant under plant-specific-expanded domain that is bounded by the MELLLA+ upper boundary. Plant-specific applications approved for operation within the MELLLA+ domain will include the plant-specific power/flow map specifying the licensed domain in the COLR.

1.4 SUMMARY AND CONCLUSIONS

The following sections of this SE cover the specific impact of MELLLA+ on the principle review topics and identify the scope of analyses that will be provided in the M+SAR. Section 12 of this SE delineates the limitations and conditions associated with the fuel-dependent analyses for operation at the proposed MELLLA+ conditions.

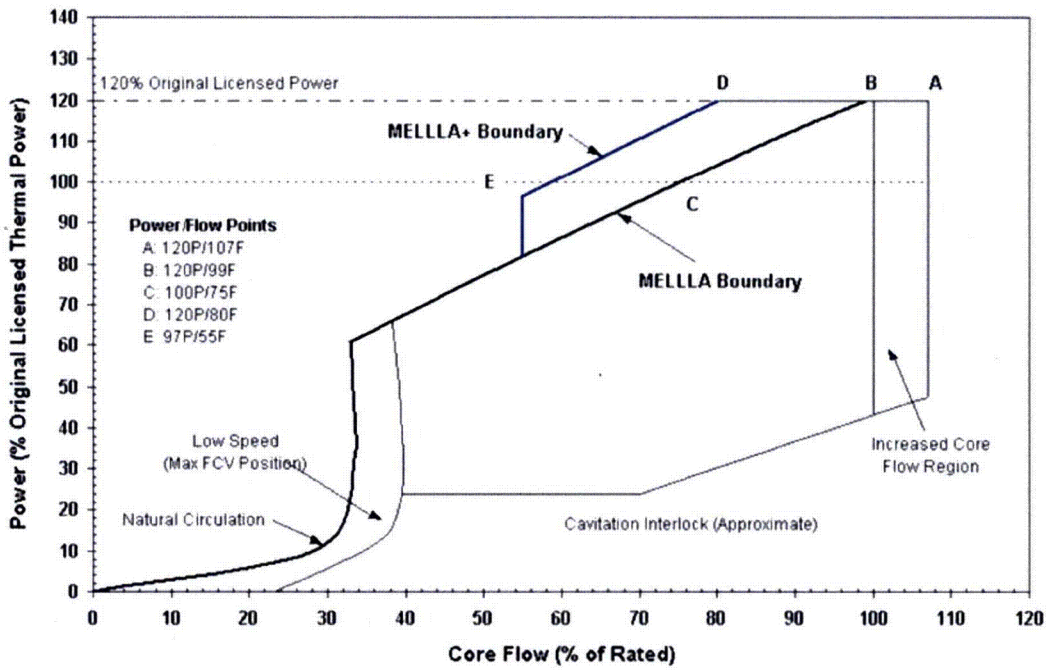

Figure 1-1 MELLLA+ Operating Range Power/Flow Map

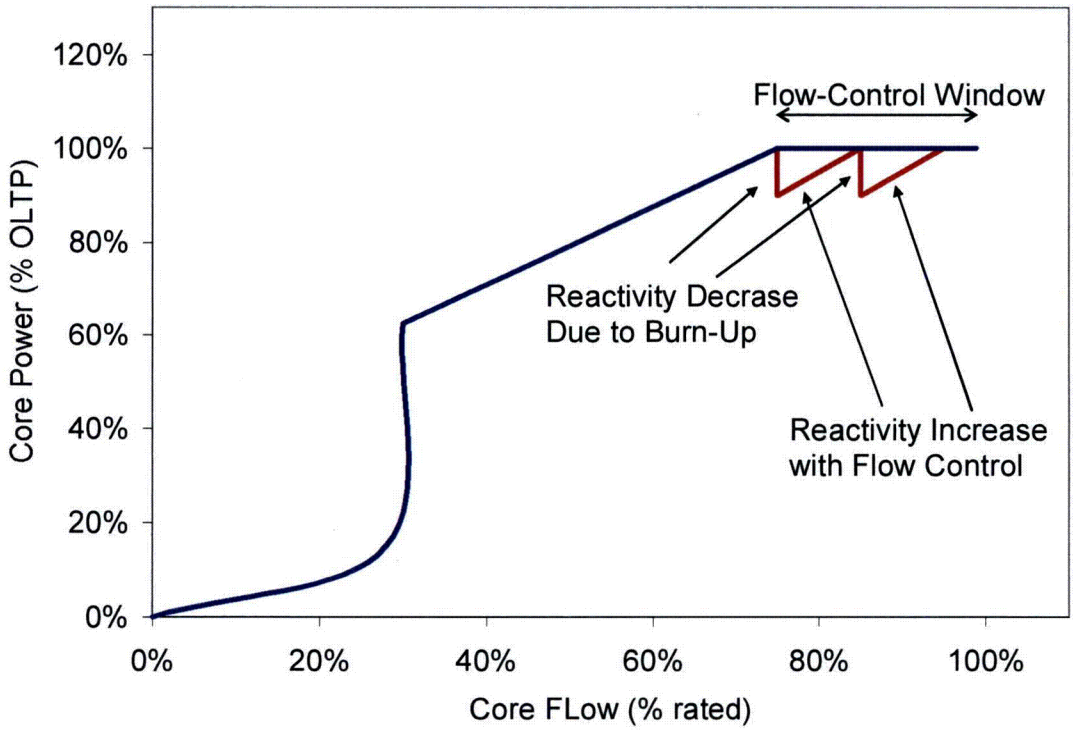

Figure 1-2 Illustration of flow control of reactivity decreased due to burn-up

2.0 REACTOR CORE AND FUEL PERFORMANCE

This section provides the NRC staff review of the reactor core and fuel performance. Table 2-1 lists the specific topics and the corresponding NRC staff dispositions for plant-specific application.

Table 2-1 Reactor Core and Fuel Performance Topics

Section	Title	[[	
2.1	Fuel Design and Operation		
2.2	Thermal Limit Assessment		
2.3	Reactivity Characteristics		
2.4	Stability		]]

Plant-specific evaluations will be reported in the plant-specific submittal consistent with the applicable limitations. The applicability of the generic assessments for a plant-specific application will be evaluated and the plant-specific submittal will either document the confirmation of the generic assessment or provide a plant-specific evaluation if the generic applicability assessment does not apply.

2.1 FUEL DESIGN AND OPERATION

The use of NRC-approved fuel design acceptance criteria and analysis methodologies assures that the fuel performs in a manner that is consistent with the NUREG-0800, "Standard Review Plan" Sections 4.2 and 4.3 and the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, applicable GDC.

Fuel is designed to ensure that:

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

2.1.1 Assessment

Fuel design limits are established for all new fuel product line designs as a part of the fuel introduction based on the NRC-approved GESTAR II approach. Changes in fuel product line designs are not allowed to be part of a MELLLA+ application (i.e., they must be reviewed and approved in a separate application) and there are no changes to fuel design limits required for MELLLA+ application. In general, the fuel design limits are evaluated on a [[ ]] as a part of the reload licensing process to ensure that the criteria for fuel design limits are met. Certain MELLLA+ effects relevant to satisfying the above criteria are

discussed in sections throughout this SE, including thermal limits (Section 2.2), stability (Section 2.4), ECCS-LOCA (Section 4.2 and 4.3), AOOs (Section 9.1), and ATWS (Section 9.3.1).

2.1.1.1 Core Void Distribution

For the proposed MELLLA+ operation, plants will operate at EPU power levels at CFs as low as 80 percent of rated. This leads to higher bundle power-to-flow ratio and changes in the core power and void (axial and radial) distribution that may challenge the margins to the fuel design limits. In the review of the NEDC-33173P (Reference 37), the NRC staff identified concerns with applicability of existing methods to EPU and MELLLA+ conditions, particularly with respect to their applicability to higher in-channel and bypass void conditions.

Table 2-1 and Table 5-1 of NEDC-33173P (shown here in Table 2-2 and Table 2-3) compare the hot channel exit void fractions and bypass void fractions respectively, at different reactor operating domains. These results show that BWR operation at the MELLLA+ statepoints results in higher void fraction in the channel and in the bypass, compared to pre-MELLLA+ conditions.

Table 2-2 Exit Void Fraction

Plant / Parameter Hot Channel	Power (%OLTP)/Core Flow (%rated)	Exit Voids
[[		
		]]

Table 2-3 Bypass Void Fractions Calculated for Different Reactor Operating Domains

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					]]

The NRC staff concludes that implementation of MELLLA+ will result in operation outside the current experience base. In LTR NEDC-33173P, GHNE evaluated the impact of operation at higher void conditions characteristic of EPU and MELLLA+ operation on all of its licensing analytical methods. LTR NEDC-33173P was reviewed and approval by the NRC staff is pending with a number of limitations. These limitations are applicable to the present review of NEDC-33006P LTR.

2.1.2 Conclusion

The applicability of the generic fuel design and operation assessments presented in LTR NEDC-33006P will be confirmed and documented in plant-specific requests to implement MELLLA+. If they cannot be confirmed, then the licensee must provide a plant-specific

evaluation documented in the plant-specific M+SAR. LTR NEDC-33006P proposes to defer the MELLLA+ cycle-specific core design and associated safety analyses to the reload analysis. In order for the NRC staff to be able to adequately evaluate the effect of MELLLA+ core designs, the applicant shall provide the plant-specific thermal limits assessment and transient analysis results. Considering the timing requirements to support the reload, the fuel- and cycle-dependent analyses including the plant-specific thermal limits assessment may be submitted by supplementing the initial M+SAR. In addition, the SRLR for the initial MELLLA+ implementation cycle shall be submitted to the NRC staff.

The approach described in the LTR NEDC-33006P is acceptable to the NRC staff, with satisfactory compliance to the applicable limitations and conditions.

2.2 THERMAL LIMITS ASSESSMENT

The regulation at 10 CFR Part 50, Appendix A, GDC-10, "Reactor design," 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).

2.2.1 Assessment

The effect of the MELLLA+ 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 below.

2.2.1.1 SLMCPR

LTR NEDC-33006P states that the impact of MELLLA+ on the safety limit MCPR (SLMCPR) is about [[ ]] However, for operation in the MELLLA+ domains, bundles will be operating at high bundle power/flow conditions, with the upper part of the maximum powered bundles operating at high void conditions. The control rod density would be different resulting in different axial power distribution than the EPU statepoint. Therefore, for operation at the MELLLA domain, the core thermal-hydraulic conditions will differ from rated EPU conditions, with the maximum powered bundles operating with higher void fractions, where the MCPR response is more limiting.

In the general GHNE SLMCPR methodology, the base core thermal-hydraulic conditions are established at different exposure points. The key parameters (e.g., power, CF, feedwater flow, etc.) that are important to the SLMCPR response are perturbed according to the corresponding uncertainties. The SLMCPR value that meets the 0.1 acceptance criteria, where 99.9 percent of the fuel rods avoid BT, establishes the SLMCPR limit. While the SLMCPR analysis methodology is complex and involves statistical treatments, overall, the SLMCPR methodology assumes that the rated core thermal-hydraulic condition, perturbed at higher uncertainties for reduced flow conditions, yields the most conservative SLMCPR value. A similar methodology is employed in the SLO conditions, in which the base rated thermal-hydraulic conditions are perturbed with higher CF uncertainties.

The methodology conclusions, in part, are drawn from sensitivity analyses that examined the changes in the SLMCPR with changes in the dominant parameters that affect the safety limit

such as the feedwater flow, total CF, channel flow area, and FWT. The SLMCPR study shows the relative sensitivity of the SLMCPR with increases in these key parameters, [[ ]] (see Figure 2-5 of this SE).

Since MELLLA+ domain involves operation at EPU power levels at reduced CF conditions, the NRC staff requested in RAI 17 that GE demonstrate why SLMCPR calculations based on the thermal-hydraulic conditions at the MELLLA+ statepoints (120 percent P/ 80 percent CF; 100 percent P/ 55 percent CF) will not result in a higher SLMCPR response than the rated condition. In MFN 04-020 (Reference 28), GHNE provided justifications that the SLMCPR methodology, as applied, results in the most bounding value. However, SLMCPR sensitivity analyses audited by the NRC staff showed that the SLMCPR value at the 55 percent CF statepoint was most limiting. In addition, SLMCPR values calculated at the minimum CF statepoint for BWRs operated at 105 percent power at minimum CF were also higher than the rated SLMCPR value.

Subsequently, GHNE issued a Part 21 evaluation documented in MFN 04-081 (Reference 39). The Part 21 evaluation stated that the power distribution, resulting from operation at the reduced flow conditions, could yield SLMCPR values that bound the rated SLMCPR value. GHNE revised its SLMCPR methodology, including calculation of the SLMCPR at minimum CF in the licensing process. The calculated SLMCPR at the minimum CF statepoint (OLTP/75 percent CF or 105 percent P/82 percent CF) for several BWRs resulted in a higher SLMCPR value than at the rated conditions. The current GHNE SLMCPR applies higher off-rated CF uncertainty for non-rated conditions.

In the revised RAI 17 response provided in MFN-07-041 (Reference 32), GHNE proposes reducing the CF uncertainty applied to the MELLLA+ statepoints (120 percent P/ 80 percent ; OLTP/ 55 percent), which will then decrease the SLMCPR value. Based on the uncertainties associated with different CF ranges, GHNE determines the values associated with the MELLLA+ statepoints and justifies applying the lower uncertainty values. Figure 2-6 of this SE shows the CF and the feedwater flow uncertainties for different CFs. SLMCPR calculations based on the reduced CF statepoint result in lower SLMCPR values, then previously computed at the reduced CF statepoints.

The NRC staff considered GHNE's proposal to apply lower CF uncertainties than is currently applied in the revised SLMCPR methodology discussed above. The SLMCPR methodology still assumes rated core thermal-hydraulic base conditions for SLO and applies higher CF uncertainty. For the MELLLA minimum CF and the proposed MELLLA+ minimum CF, the base core thermal-hydraulic conditions will be predicted, but the higher off-rated SLO CF uncertainty is applied.

The NRC staff finds that the proposal to apply graded CF uncertainty is not acceptable, because the power distribution uncertainty applied to the minimum CF statepoints is assumed to be the same as the rated conditions. Any changes in the power distribution uncertainties at the top part of the fuel bundles for operation at the reduced CF statepoints cannot be benchmarked through gamma scan. The reason is that the gamma scans capture the last 60 days of reactor operation, when the reactor will be operating at the rated conditions.

As discussed in NEDC-33173P (Reference 37), the core-wide power distribution uncertainties based on TIP comparisons show that the axial and nodal power distribution uncertainties change with core power-to-flow ratio. This assessment is based on core-wide power-to-flow ratios and calculated power distribution uncertainties for a given cycle statepoint. Therefore, the specific changes in the bundle axial and nodal power distributions with higher power/flow conditions cannot be assessed or validated.

Since the rated power distribution uncertainties (e.g., σ_{peak} and σ_{bundle}) cannot be directly validated for the higher bundle power/flow conditions characteristic of the operation at the MELLLA+ reduced flow conditions, the higher CF uncertainty provides confidence in the calculated SLMCPR value.

In addition, the MELLLA+ operation will represent operation outside the current experience base. GHNE is expected to update its SLMCPR methodology for the proposed operating strategies. The SLMCPR submittal will update the NRC-approved SLMCPR methodology specified in References 40, 41, 42, and 43. The SLMCPR submittal will also address specific topics relevant to the operation at the expanded operating domains, such as the limiting control rod patterns assumed in modeling the base core thermal-hydraulic conditions at the reduced CF conditions.

Therefore, the NRC staff concludes that the currently used SLMCPR methodology, based on the Part 21 report applies until the NRC staff reviews and approves the updated SLMCPR methodology. The NRC staff finds that with consistent application of the increased CF uncertainty at the MELLLA+ upper boundary statepoints, the application of the rated power distribution uncertainties (σ_{peak} and σ_{bundle}) is acceptable.

SLMCPR Statepoints and CF Uncertainty Limitation

Until such time when the SLMCPR methodology (References 40 and 41) for off-rated SLMCPR calculation is approved by the staff for MELLLA+ operation, the SLMCPR will be calculated at the rated statepoint (120 percent P/100 percent CF), the plant-specific minimum CF statepoint (e.g., 120 percent P/80 percent CF), and at the 100 percent OLTP at 55 percent CF statepoint. The currently approved off-rated CF uncertainty will be used for the minimum CF and 55 percent CF statepoints. The uncertainty must be consistent with the CF uncertainty currently applied to the SLO operation or as NRC-approved for MELLLA+ operation. The calculated values will be documented in the SRLR.

Table 2-4 below shows the statepoints for which the SLMCPR will be calculated. If a specific plant, can achieve increased CF, the SLMCPR will be calculated for the EPU power level at the ICF conditions.

Section 5.1.1.5, "Additional Review Topics," of this SE contains additional bases for using the higher CF uncertainties.

Table 2-4 SLMCPR Analysis Conditions for MELLLA+

Power (Percent Rated)[1]	Flow (Percent Rated)
120 percent	100 percent
120 percent	80 percent[2]
97 percent	55 percent

[1] or corresponding maximum allowable power level at the specified CF shall be used.

[2] or minimum MELLLA+ CF submitted in the license application.

The MELLLA+ will not change the requirement to perform the SLMCPR analysis for each reload core, reflecting the actual plant core-loading pattern. This will be based on the NRC-approved GESTAR II methodology and the cycle-specific SLMCPR will be determined. The licensee will submit an amendment request if the cycle-specific SLMCPR values exceed the TS.

Based on the approach discussed, the NRC staff finds the SLMCPR methodology is acceptable for operation at MELLLA+ domain.

2.2.1.2 Operating Limit MCPR (OLMCPR)

The OLMCPR is calculated by adding the change in the MCPR, due to the limiting AOO event, to the SLMCPR for non-TRACG methods. The OLMCPR is determined on a cycle-specific basis from the results of the reload transient analyses, as described in GESTAR II. The cycle-specific analysis results are documented in the SRLR and included in the COLR. MELLLA+ does not change the approach used to determine this limit. The MELLLA+ impact on AOO change in CPR (ΔCPR), including off-rated limits is discussed in Section 9.1 of this SE.

2.2.1.3 MAPLHGR and LHGR

The MAPLHGR limits ensure that the plant does not exceed regulatory limits established in 10 CFR 50.46. [[

]] MELLLA+ does not change the approach used to determine this limit. The MELLLA+ impact on ECCS performance is discussed in Section 4.3 of this SE.

The LHGR limits ensure that the plant does not exceed the fuel thermal-mechanical design limits. The steady state LHGR limit is determined by the fuel rod thermal-mechanical design. [[

]] The NRC staff draft SE of NEDC-33173P (Reference 38) contains additional assessments and limitations associated with the transient LHGR limit response for the EPU/MELLLA+ operation. These limitations are applicable until a later version of NEDC-33173P is approved.

2.2.1.4 Methods Assessment

In the review of the NEDC-33173P (Reference 37), the NRC staff identified concerns with the applicability of existing methods to EPU and MELLLA+ conditions, particularly with respect to their applicability to higher in-channel and bypass void conditions.

In LTR NEDC-33173P, GHNE evaluated the impact of operation at higher void conditions characteristic of EPU and MELLLA+ operation on all of its licensing analytical methods. LTR NEDC-33173P was reviewed and approval is pending by the NRC staff with a number of limitations. These limitations are applicable to the present review of LTR NEDC-33006P.

2.2.2 Conclusion

The applicability of the generic assessments presented in LTR NEDC-33006P will be confirmed and documented in plant-specific requests to implement MELLLA+. If they cannot be confirmed, then the licensee must provide a plant-specific evaluation in the plant-specific M+SAR.

In order for the NRC staff to be able to adequately evaluate the effect of MELLLA+ on core designs, the plant-specific application shall provide the plant-specific thermal limits assessment and transient analysis results. Considering the timing requirements to support the reload, the fuel and cycle dependent analyses, including the plant-specific thermal limits assessment, may be submitted by supplementing the initial M+SAR. In addition, the SRLR for the initial MELLLA+

implementation cycle shall be submitted to the NRC staff. The NRC staff finds the thermal limits approach discussed above acceptable.

2.3 REACTIVITY CHARACTERISTICS

The regulation at 10 CFR Part 50, Appendix A, GDC-26, "Reactivity Control System Redundancy and Capability," requires that:

> Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including AOOs, and with appropriate margin for malfunctions such as stuck rods, SAFDLs are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

2.3.1 Assessment

The effect of MELLLA+ on the minimum SDM and hot excess reactivity is discussed in LTR NEDC-33006P. The topics addressed in this evaluation are 1) hot excess reactivity, and 2) SDM.

MELLLA+ core design may affect the hot excess core reactivity and may also affect operating SDMs. Higher core average void fraction, higher Pu production, increased hot reactivity later in the operational cycle, decreased hot-to-cold reactivity differences, and smaller cold SDMs may result from cores designed for operation with the MELLLA+ operating range expansion. 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. LTR NEDC-33173P and associated NRC staff SE provide additional discussion on the SDM.

2.3.2 Conclusion

The applicability of the generic assessment presented in LTR NEDC-33006P will be confirmed in the licensee's plant-specific submittal. If they cannot be confirmed, then the licensee must provide a plant-specific evaluation in the plant-specific M+SAR.

2.4 STABILITY

Coupled neutronic-thermal-hydraulic instabilities, also known as density-wave instabilities, are a safety concern for BWRs. There are three recognized modes of density-wave instability: (1) core-wide instability (all the core channels oscillate in phase); (2) regional instability (half the core channels oscillate out-of-phase with the other half); and (3) single-channel flow instability (the flow in a single channel oscillates with little or no power oscillations).

Certain instability events can lead to unacceptable consequences to the fuel if the reactor is not shut down on time. Specifically, for the density-wave regional stability mode, the original reactor protection system could not guarantee a timely shutdown because the APRM signal averages the positive and negative sides of the power oscillation, so the oscillation amplitude sensed by

the APRM is significantly smaller than the actual power oscillation experienced by the channels. Methodologies for resolving BWR core-stability issues are presented in GE LTR NEDO-31960A, "BWR Owner's Group Long-Term Stability Solutions Licensing Methodology," May 1991, along with its supplement, NEDO-31960A Supplement 1, "BWR Owner's Group Long-Term Stability Solutions Licensing Methodology," March 1992, which were approved by the NRC. These reports provide LTSs to BWR stability issues as well as methodologies developed to support the design of systems needed to ensure that plants comply with GDC-10, "Reactor design," and GDC-12, "Suppression of reactor power oscillations."

The NRC's acceptance criteria are based on the following:

1. GDC-12 insofar as it requires that oscillations are either not possible or can be reliably and readily detected and suppressed;

2. GDC-10 insofar as it requires that the reactor coolant system be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs and instability events;

3. Generic Letter (GL) 94-02 insofar as it states that all reactors must install a stability LTS that ensures that GDC-10 and GDC-12 are satisfied.

2.4.1 Stability Assessment

MELLLA+ operation reduces the stability margin of the reactor, when compared to NRC-approved operating envelopes (e.g., OLTP, or MELLLA). Thus, reactors operating under MELLLA+ will be more likely to experience instability events.

Figure 2-1 of this SE illustrates the reasons for the decrease in stability margin induced by MELLLA+ operation. This figure represents a typical reactor power-to-flow map, with a typical stability boundary line. Operation in the instability region (to the left of or above the stability boundary) can result in unstable power oscillations. As seen in this figure, when operating in the OLTP 100 percent rod line (e.g., at 100 percent OLTP, 100 percent flow), the reactor may or may not become unstable following a 2RPT. When operating on the MELLLA rod line (e.g., 120 percent OLTP, 99 percent CF), the reactor enters the instability region following a 2RPT; thus, the possibility of instability is high. When operating on the MELLLA+ line (e.g., 120 percent OLTP, 80 percent CF), the final operating point following a 2RPT is so far into the instability region that unstable power oscillations are essentially guaranteed.

Numerical evaluations of the impact of MELLLA+ operation on stability using the TRACG code indicate that instabilities will develop in a very short time following a 2RPT (of the order of 10 seconds), and they will result in CPR violations in less than one minute. Figure 2-2 and Figure 2-3 of this SE show TRACG simulations of a 2RPT from MELLLA+ conditions.

Therefore, the NRC staff concludes that manual operator actions are not adequate to control the consequences of instabilities when operating in the MELLLA+ domain. Given the fast nature and rapid consequences of these transients under MELLLA+ conditions, the stability LTS used on these plants must be reviewed for applicability to these harsher conditions. LTS that were reviewed and approved for OLTP may not automatically be applicable to these new operating conditions.

In the past, stability LTS implementations have relied on the Interim Corrective Actions (ICAs) in the event that the primary LTS is declared inoperable. ICAs have been in place since the early 1990's and rely on operator actions to recognize and suppress the oscillations should they

occur. Given the fast nature of the instability events under MELLLA+ conditions, operator actions are not an acceptable method to detect and suppress oscillations. Therefore, stability LTS options must include an approved backup stability solution to operate in the MELLLA+ region when the primary LTS is declared inoperable. Alternatively, reactors may have a TS requirement to exit the MELLLA+ region when the primary LTS is not operable.

Stability Limitation

Manual operator actions are not adequate to control the consequences of instabilities when operating in the MELLLA+ domain. If the primary stability protection system is declared inoperable, a non-manual NRC-approved backup protection system must be provided, or the reactor core must be operated below a NRC-approved backup stability boundary specifically approved for MELLLA+ operation for the stability option employed.

GHNE has evaluated the applicability of Solution III (i.e., detect and suppress solution (DSS)) to MELLLA+ operation. GHNE has concluded that implementation of Solution III to MELLLA+ operation would result in prohibitively small scram setpoints, which would have adverse effect on normal operation. The NRC staff concurs with GHNE's evaluation that Solution III is not directly applicable to MELLLA+ operation.

GHNE has proposed a new stability LTS, Detect and Suppress Solution - Confirmation Density (DSS-CD), for use under MELLLA+ conditions. LTR, NEDC-33075P, "General Electric Boiling Water Reactor Detect and Suppress Solution - Confirmation Density", July 2004 (Reference 45), has been reviewed and approved by the NRC staff (Reference 46).

Other stability LTSs (e.g., Solution E1A, II, or ID) have not been evaluated or reviewed for MELLLA+ operation. Plant-specific MELLLA+ applications will require an evaluation of these stability LTSs.

2.4.1.1 LTR NEDC-33075P DSS-CD Description

Section 3 of NEDC-33075P (Reference 45) describes in detail the DSS-CD methodology. In summary, DSS-CD is based on the approved Solution III, and it shares most of its features. There are only two major differences between Solution III and DSS-CD:

1. DSS-CD does not require an amplitude setpoint to trigger scram actuation if the period-based detection algorithm (PBDA) identifies an instability event. With DSS-CD implemented, the reactor will trip automatically if a coherent oscillation of any amplitude (e.g., only 1 percent) is identified. Therefore, DSS-CD does not rely on generic correlations like Delta CPR over Initial CPR vs. Oscillation Magnitude (DIVOM) or cycle-specific calculations.

2. To prevent spurious scrams, DSS-CD requires PBDA confirmations of a significant number of OPRM cells; thus the name "density of confirmations". The confirmation density algorithm (CDA) is relatively complex to cover all possibilities of combinations of failed and unresponsive OPRM cells, but under most conditions, if at least five OPRM cells confirm the instability, the reactor will scram.

Other features of the DSS-CD methodology include:

1. DSS-CD maintains the defense-in-depth algorithms that were approved for Solution III: the PBDA, the amplitude based algorithm (ABA), and the growth rate algorithm (GRA). The ABA and GRA algorithms remain unchanged from the approved solution and provide

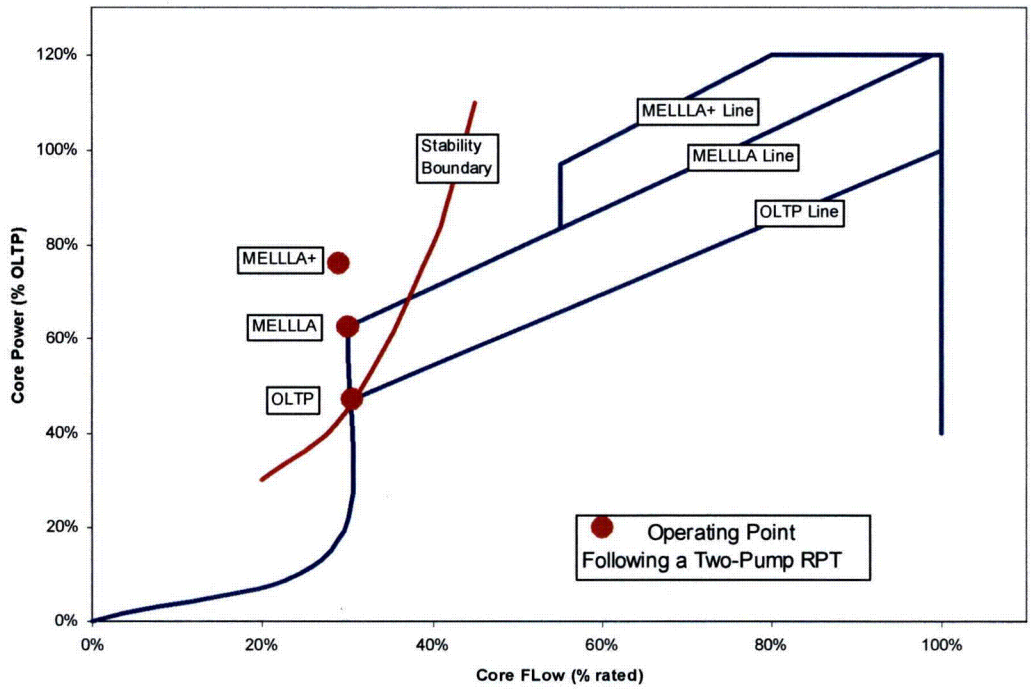

Figure 2-1 Illustration of the power-flow map, showing that MELLLA+ operation will result in deeper penetration inside the instability region following a two-pump RPT

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Figure 2-2 Hot channel power following a 2RPT from MELLLA+ shows unstable oscillations within 10 to 15 seconds of the pump trip (Fig 4.3 of NEDC-33075P)

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Figure 2-3 Hot channel CPR following a 2RPT from MELLLA+ shows CPR less than 1.0 within 45 seconds of the pump trip (Fig 4.5 of NEDC-33075P)

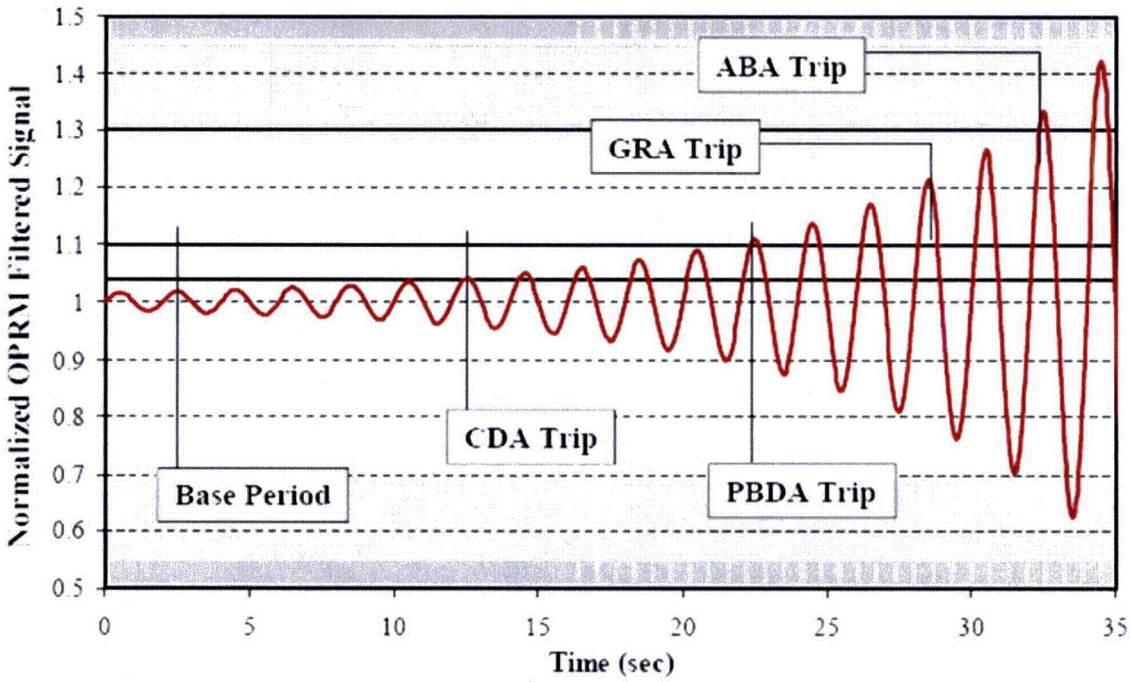

Figure 2-4 Illustration of DSS/CD Defense in Depth Algorithms

defense in depth in the unlikely event that the CDA algorithm fails to detect the instability due to unforeseen situations.

2. PBDA was the primary algorithm in Solution III, and it is retained in DSS-CD with fixed parameter settings documented in Table 3-4 of NEDC-33075P (Reference 45). PBDA will provide a scram if a single OPRM (in each protection system channel) provides 15 confirmations with amplitude greater than 110 percent. PBDA, thus, provides defense in depth just in case the CDA fails in an unexpected mode.

3. DSS-CD can be implemented as a software change using the existing GHNE NUMAC hardware used currently for Solution III. This review does not address implementation with non-GHNE hardware.

4. In addition to the DSS-CD algorithm, NEDC-33075P (Reference 45), describes a backup stability protection (BSP) methodology. The BSP is intended to provide SLMCPR protection if the regular DSS-CD is declared inoperable. With BSP, the DSS-CD methodology attempts to incorporate the lessons learned from recent Part 21 notifications, when the primary stability protection system is declared inoperable.

Figure 2-4 of this SE illustrates the operation of the main DSS-CD algorithm (CDA) and the defense-in-depth algorithms (PBDA, GRA, and ABA). The defense-in-depth algorithm would only be required in case the CDA algorithm failed for an unforeseen reason. They are armed when the oscillation amplitude reaches either 10 percent (PBDA and GRA) or 30 percent (ABA).

The BSP is described in Section 7 of NEDC-33075P (Reference 45) and consists of three different options: (a) manual BSP, (b) automated BSP, and (c) BSP boundary. All three BSP options define cycle-specific exclusion regions, which are defined in the COLR. In the automated BSP option, the scram is performed automatically by the DSS-CD hardware. In the manual BSP option, the scram is enforced administratively. The BSP boundary option limits high power operation when DSS-CD is not operable to ensure that a two-pump RPT transient will not result in unstable conditions inside the exclusion region.

The BSP methodology is an integral part of DSS-CD for operation in the MELLLA+ domain if the DSS-CD option is declared inoperable. However, the applicability of the BSP is not limited to DSS-CD. It may also be used in plants with other LTSs to replace the current ICAs. The main advantage of BSP over ICAs is that BSP requires plant- and cycle-specific stability exclusion regions; therefore, more stable plants have smaller exclusion regions and less stable plants have larger regions. ICAs are generic in nature and treat all plants by the same norm. They are based mostly on historical plant operating experience, which may or may not be applicable to new fuels and operating strategies that include high power densities with flat power distributions. By requiring plant- and cycle-specific region calculations, the BSP methodology guarantees that the stability regions are up to date for each particular core loading and operating strategy.

The DSS-CD methodology has been reviewed and approved for MELLLA+ application using TRACG as the analysis tool (Reference 47) by the NRC staff (References 46 and 48).

2.4.2 Stability Conclusion

As discussed above, the NRC staff review concludes that manual operator actions are not adequate to control the consequences of instabilities when operating in the MELLLA+ domain. The stability approach described in LTR NEDC-33006P is acceptable to the NRC staff given the limitation in Section 2.4.1 of this SE.

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Figure 2-5 Four Dominant SLMCPR Sensitivities for a Factor Change in the Generic GETAB Uncertainty Value

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Figure 2-6 Total Core Flow and Feedwater Flow Uncertainties for BWRs 4/5/6

3.0 REACTOR COOLANT AND CONNECTED SYSTEMS

This section provides the NRC staff review of the reactor coolant and connected systems. Table 3-1 lists the specific topics and the corresponding NRC staff dispositions for plant-specific application.

Table 3-1 Reactor Coolant and Connected Systems Topics

Section	Title	[[	
3.1	Nuclear System Pressure Relief/Overpressure Protection		
3.2	Reactor Vessel		
3.3	Reactor Coolant Pressure Boundary (RCPB) Piping		
3.4	Main Steam Line Flow Restrictors		
3.5	Main Steam Line Valves		
3.6	Reactor Core Isolation Cooling/Isolation Condenser (IC)		
3.7	Main Steam Line Flow Restrictors		
3.8	Main Steam Isolation Valves		
3.9	Reactor Core Isolation Cooling/Isolation Condenser		
3.10	Residual Heat Removal System		
3.11	Reactor Water Cleanup System		]]

Plant-specific evaluations will be reported in the plant-specific submittal consistent with the applicable limitations. The applicability of the generic assessments for a plant-specific application will be evaluated and the plant-specific submittal will either document the confirmation of the generic assessment or provide a plant-specific evaluation if the generic applicability assessment does not apply.

3.1 NUCLEAR SYSTEM PRESSURE RELIEF AND OVERPRESSURE PROTECTION

The topics addressed in this evaluation are:

Table 3-2 Pressure Relief Topics Addressed

Topic	MELLLA+ Effect	Disposition
Overpressure Relief Capacity	None	[[ ]]

The relief and safety valves and the reactor protection system provide overpressure protection for the reactor coolant pressure boundary (RCPB) during power operation. The NRC staff's review covered relief and safety valves on the main steam lines and piping from these valves to the suppression pool. The NRC's acceptance criteria are based on (1) draft GDC-9, insofar as it requires that the RCPB be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime; and (2) draft GDC-33, -34, and -35, insofar as they require that the RCPB be designed to assure that it behaves in a non-brittle manner and that the probability of rapidly propagating type failures is

minimized. Specific review criteria are contained in SRP Section 5.2.2 and other guidance provided in Matrix 8 Table of RS-001, "Review Standard for Extended Power Uprates," (Reference 49). The pressure relief systems provide reactor overpressure protection for the NSSS to prevent failure of the nuclear system pressure boundary and uncontrolled release of fission products, during abnormal operational transients, the ASME Upset overpressure protection event, and postulated ATWS events. Section 9.3.1 of this SE evaluates the ATWS response for operation at the MELLLA+ operating domain.

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

3.1.1 Assessment

The SRV setpoints are established to provide the reactor overpressure protection function, while ensuring that there is adequate margin between the reactor operating pressure and the SRV actuation setpoints to prevent unnecessary SRV actuations during normal plant maneuvers. No changes in the pressure relief system or SRV setpoints are expected for MELLLA+. The abnormal operational transients, the ASME overpressure analyses, and the ATWS response MELLLA+ evaluations are performed using the existing SRV setpoint tolerances.

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]]. The M+SAR will justify the basis that the limiting ASME overpressure event changed. The bounding ASME overpressure event will be performed at both the minimum and maximum flow rate statepoints for the plant-specific MELLLA+ applications.

The plant-specific M+SAR will:

1. document the modifications in the existing licensing basis analysis to calculate the peak vessel pressure for ASME overpressure (e.g., increase in the number of SRVs credited),

2. demonstrate that the SRV tolerances assumed in the ASME Overpressure calculation is based on the actual SRV performances using NRC-approved or accepted uncertainty and tolerance treatment, and

3. document that the assumptions and code inputs for the ASME Overpressure calculation are consistent with the existing licensing basis.

The M+SAR will include a plant-specific evaluation of the limiting ASME overpressure event to confirm the adequacy of the pressure relief system for MELLLA+ conditions. The limiting ASME overpressure event is analyzed on cycle- and core-configuration specific conditions during the standard reload process and the results are documented in the SRLR.

3.1.2 Conclusion

The ASME overpressure analysis will be performed at rated CF, ICF if achievable, and the minimum CF statepoints. The plant-specific MELLLA+ application will demonstrate that the

SRV tolerances used in the ASME overpressure analyses are based on the actual plant SRV performance, in terms of SRV tolerances and SRV out of service options.

3.2 REACTOR VESSEL

3.2.1 Fracture Toughness

LTR NEDC-33006P stated that the MELLLA+ operating range expansion may result in a higher operating neutron flux at the vessel wall due to the increased void fraction in the core, and a consequent increase of the integrated flux over time (fluence). This increase is small and will have a minor effect on the vessel. LTR NEDC-33006P also stated that any licensee seeking use of the MELLLA+ operating range will need to provide a plant-specific evaluation of the reactor pressure vessel (RPV) fluence and fracture toughness. Specifically, the licensee will 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 and decrease in USE values for a given material will be calculated in accordance with Regulatory Guide (RG) 1.99, Revision 2 (Reference 50). With regard to evaluating the effect of MELLLA+ on the RPV ART values and pressure-temperature (P-T) limits, GE stated that, for the case where the plant's P-T limit curves are beltline limited and the ART increases, then new P-T curves will be required. 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 system hydrostatic tests, to which the pressure boundary may be subjected to over its service life.

With regard to evaluating the effect of MELLLA+ on USE, GE stated that the values for the vessel materials at the end of life must meet the 50 ft-lb criterion 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 LTR, GE stated that it performed a generic EMA for the RPV materials of the U.S. BWR fleet in Reference 3 of LTR NEDC-33006P, which was approved by the NRC in an SE to Gulf States Utilities Company dated December 8, 1993. However, GE concluded that a plant-specific evaluation will be required to demonstrate that the RPV materials would continue to meet the limits for the EMA.

The NRC staff concurs that applicants proposing to use MELLLA+ will need to perform revised plant-specific neutron fluence assessments for the RPV materials and that those assessments must be performed in accordance with the most up-to-date 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 10 CFR 50.60(a) which 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. The regulation at 10 CFR 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. The regulation at 10 CFR 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's TSs.

Therefore, licensees seeking to use LTR NEDC-33006P as their basis for MELLLA+ license amendments will have to evaluate all beltline materials for ART and USE based on the plant-specific MELLLA+ based fluence values. The ART is to be evaluated for beltline materials

including any materials that are added to the beltline list. The current plant-specific P-T limit curves are evaluated relative to the change in ART. If the change in ART results in new and bounding P-T limit curves, GE will recommend that the P-T curves be revised. Pursuant to 10 CFR 50.90, if this occurs, the licensee must submit a license amendment request for NRC approval of the new limiting P-T curves.

In addition, the licensee must demonstrate that either the USE values for all beltline materials, as determined from the MELLLA+ based fluence levels, will remain above 50 ft-lb throughout the licensed life of the plant, or that GE staff-approved generic EMA analysis, as provided in Reference 3 of LTR NEDC-33006P remains bounding for their MELLLA+ 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 LTR NEDC-33006P.

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

The NRC staff's review of the methods in LTR NEDC-33006P, Section 3.2.1, indicates that the methods and analyses in the LTR are generally acceptable. The NRC staff finds that, with the addition of the limitation below, GE has provided adequate specific direction to the BWR licensees for assessing the impact of MELLLA+ on their facilities. Therefore, the NRC staff concludes that a licensee's adherence to the requirements of LTR NEDC-33006P and the completion of the limitation below, will facilitate future NRC staff reviews of MELLLA+ licensing amendment requests. This LTR may be used as a reference for implementing MELLLA+, concerning these sections in a license amendment for GE designed BWRs to the extent specified and under the limitations delineated in this SE.

Fluence Methodology and Fracture Toughness Limitation

The applicant is to provide a plant-specific evaluation of the MELLLA+ RPV fluence using the most up-to-date NRC-approved fluence methodology. This fluence will then be used to provide a plant-specific evaluation of the RPV fracture toughness in accordance with RG 1.99, Revision 2.

3.2.2 Reactor Vessel Structural Evaluation

The reactor vessel components will not be affected by the proposed expansion of the power/flow map because none of the parameters that affect the stress of fatigue for the reactor vessel components (i.e., reactor operating pressure, feedwater flow or steam flow rate, or other applicable mechanical loads) will be changed or increased.

3.3 REACTOR INTERNALS

3.3.1 Reactor Internal Pressure Differences

The reactor vessel pressure differences will not be affected by the proposed expansion of the power/flow map because none of the parameters that affect reactor vessel pressure differences (i.e., core exit steam flow, operating pressure, and feedwater flow and steam flow) will be changed.

General Electric Company (GE) considered the faulted acoustic and flow induced loads in the RPV annulus resulting from the recirculation line break loss-of-coolant accident (LOCA) in its evaluation and stated that the conclusion depends on the minimum flow in the MELLLA+ region and the lowest feedwater temperature evaluated in the plant design basis. Therefore, since there is a possible small increase for some components from MELLLA+ operation, the NRC staff will evaluate the affect on a plant-specific basis. The NRC staff finds this to be acceptable since it ensures that the safety analysis will be performed and that it will be appropriate for every plant proposing to use MELLLA+.

3.3.2 Reactor Internals Structural Evaluation

GE stated that the [[ ]] may be affected by load increases due the MELLLA+ and that the plant-specific M+SAR (M+SAR) will include structural integrity evaluation of these components for the MELLLA+ operating range expansion. Therefore, the NRC staff will evaluate the affect of MELLLA+ operation on [[ ]] on a plant-specific basis. The NRC staff finds this to be acceptable since it ensures that the safety analysis will be performed and that it will be appropriate for every plant proposing to use MELLLA+.

3.3.3 Steam Separator and Dryer Performance

The performance of the steam separator and dryer are evaluated to determine the quality of the steam leaving the reactor pressure vessel. GE stated that "the MELLLA+ flow and quality conditions may result in an increase in the moisture content of the steam leaving the RPV [reactor pressure vessel]," and that "the plant-specific M+SAR will include a discussion of the steam separator and dryer performance evaluation." Therefore, the NRC staff will evaluate the affect of MELLLA+ operation on the steam separator and dryer performance on a plant-specific basis. The NRC staff finds this to be acceptable since it ensures that the safety analysis will be performed and that it will be appropriate for every plant proposing to use MELLLA+.

3.4 FLOW INDUCED VIBRATION

The flow induced vibrations will not be affected by the proposed expansion of the power/flow map because none of the parameters that affect the flow induced vibrations (i.e., flow rate in the reactor coolant pressure boundary (RCPB) piping, RCPB piping components, and RPV internals) will be changed.

3.5 PIPING EVALUATION

3.5.1 Reactor Coolant Pressure Boundary Piping

Section 3.5.1 of LTR NEDC-33006P states that RCPB piping are required to comply with the structural requirements of ASME Code or an equivalent Code applicable at the time of construction or the governing code used in the stress analysis for a modified component. In addition, the LTR states that because there is no increase in pressure, temperature and flow rate the RCPB piping is not affected. The NRC staff agrees with this assessment for Category "A" material as defined in NUREG-0313, Revision 2 (Reference 51). However, EPU applicants must identify all other than Category "A" materials that exist in its RCPB piping and discuss the adequacy of the augmented inspection programs in light of the EPU on a plant-

specific basis. This NRC staff requirement is based on the fact that many BWR plants have other than type "A" materials installed in their RCPB piping and in some cases service induced flaws are present in the RCPB piping. The presence of service flaws induced flaws in RCPB piping does not meet the original construction Code criteria, and therefore a plant-specific evaluation is required.

The NRC staff finds that a plant-specific assessment for MELLLA+ which includes the proper inspection programs associated with RCPB piping materials other than Category "A" materials is necessary to provide assurance that degradation is promptly identified and corrected so that the RCPB piping will continue to perform in service as designed.

The NRC staff's review of the methods in LTR NEDC-33006P, Section 3.5.1, indicates that the methods and analyses in the LTR are generally acceptable. The NRC staff finds that, with the addition of the applicant limitation below, GE has provided adequate specific direction to the BWR licensees for assessing the impact of MELLLA+ on their facilities. Therefore, the NRC staff concludes that a licensee's adherence to the requirements of LTR NEDC-33006P, and the completion of the limitation below, will facilitate future NRC staff reviews of MELLLA+ licensing amendment requests. This LTR may be used as a reference for implementing MELLLA+, concerning these sections in a license amendment for GE designed BWRs to the extent specified and under the limitations delineated in this SE.

Reactor Coolant Pressure Boundary Limitation

MELLLA+ applicants must identify all other than Category "A" materials, as defined in NUREG-0313, Revision 2, that exist in its RCPB piping, and discuss the adequacy of the augmented inspection programs in light of the MELLLA+ operation on a plant-specific basis.

3.5.2 Balance of Plant (BOP) Piping

The BOP piping will not be affected by the proposed expansion of the power/flow map because none of the parameters that affect the balance of plant piping (i.e., flow, pressure, temperature, and mechanical loads) will be increased.

3.6 REACTOR RECIRCULATION SYSTEM (RRS)

The RRS will not be affected by the proposed expansion of the power/flow map, because the RRS operating conditions for MELLLA+ are within the previously approved MELLLA RRS operating range. Additionally, per the MELLLA+ LTR, single loop operation is not allowed in the MELLLA+ operating domain.

3.7 MAIN STEAM LINE FLOW RESTRICTORS

There will be no effect on [[ ]] the main steam line flow restrictor by the proposed expansion of the power/flow map, because there is no increase in the steam flow rate for the MELLLA+ operating range expansion.

3.8 MAIN STEAM ISOLATION VALVES (MSIVS)

There are no [[ ]] effects on the MSIVs by the proposed expansion of the power/flow map because there is no increase in the parameters that affect the MSIVs

(i.e., pressure, steam flow rate, and pressure drop) for the MELLLA+ operating range expansion.

3.9 REACTOR CORE ISOLATION COOLING/ISOLATION CONDENSER

The reactor core isolation cooling (RCIC) system serves as a standby source of cooling water to provide a limited decay heat removal capability whenever the main feedwater system is isolated from the reactor vessel. In addition, the RCIC system may provide decay heat removal necessary for coping with a station blackout (SBO) and ATWS. The water supply for the RCIC system comes from the condensate storage tank, with a secondary supply from the suppression pool. The NRC staff's review covered the effect of the proposed MELLLA+ on the functional capability of the system.

The NRC's acceptance criteria are based on: (1) GDC-40 and -42, insofar as they require that protection be provided for engineered safety features (ESFs) against the dynamic effects that might result from plant equipment failures, as well as the effects of a LOCA; (2) GDC-37, insofar as it requires that ESFs be provided to back up the safety provided by the core design, the RCPB, and their protective systems; (3) GDC-51 and -57, insofar as they require that piping systems penetrating containment be designed with appropriate features as necessary to protect from an accidental rupture outside containment and the capability to periodically test the operability of the isolation valves to determine if valve leakage is within acceptable limits; and (4) 10 CFR 50.63, insofar as it requires that the plant withstand and recover from a SBO of a specified duration.

3.9.1 Assessment

The RCIC system provides inventory makeup to the reactor vessel when the vessel is isolated from the normal high pressure makeup systems. For BWR/3 systems that include an isolation condenser (IC), this equipment 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 evaluation of the RCIC system, used in all BWR/4, 5 and 6 and some BWR/3 plants, is based on the ability to provide 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, such as meeting the ATWS vessel pressure requirements, to maintain the reactor water level above the top of active fuel (TAF) at the MELLLA+ conditions. The system performance must be confirmed in the plant-specific application.

Sufficient net positive suction head (NPSH) must be available for the RCIC pump for projected operation at MELLLA+. Systems using the suppression pool as the makeup source may potentially lead to cavitation concerns following an ATWS event from MELLLA+ condition. System performance relative to potential changes in the makeup source conditions must be addressed in the plant-specific application.

The IC system, used on some BWR/3 plants, provides the equivalent decay heat removal function as the RCIC system for isolation events and must satisfy the same requirements. The system performance must be confirmed in the plant-specific submittal.

3.9.2 Conclusion

The licensee's plant-specific submittal will confirm the acceptability of the system performance consistent with the surveillance test results and projected MELLLA+ conditions. Therefore, the approach described in LTR NEDC-33006P is acceptable to the NRC staff.

3.10 RESIDUAL HEAT REMOVAL (RHR) SYSTEM

The RHR system will not be affected by the proposed expansion of the power/flow map, because none of the parameters that affect the RHR system (i.e., reactor operating pressure, power, sensible heat, or decay heat) will be changed.

3.11 REACTOR WATER CLEANUP (RWCU) SYSTEM

The RWCU system will not be affected by the proposed expansion of the power/flow map, because there is no change in the pressure or fluid thermal conditions experienced by the RWCU system for the MELLLA+ operating range.

4.0 ENGINEERED SAFETY FEATURES

This section provides the NRC staff review of the ESFs. Table 4-1 lists the specific topics and the corresponding NRC staff dispositions for plant-specific application.

Table 4-1 ESF Topics

Section	Title	[[	
4.1	Containment System Performance		
4.2	ECCSs		
4.3	ECCSs Performance		
4.4	Main Control Room Atmosphere Control System		
4.5	Standby Gas Treatment System		
4.6	MSIV Leakage Control System		
4.7	Post-LOCA Combustible Gas Control System		]]

Plant-specific evaluations will be reported in the plant-specific submittal consistent with the applicable limitations of the SE approving the most recent version of LTR NEDC-33173P. The applicability of the generic assessments for a plant-specific application will be evaluated and the plant-specific submittal will either document the confirmation of the generic assessment or provide a plant-specific evaluation if the generic applicability assessment does not apply.

4.1 CONTAINMENT SYSTEM PERFORMANCE

The following requirements of 10 CFR Part 50, Appendix A, GDCs are pertinent to aspects of MELLLA+ related to the primary containment:

GDC-4, "Environmental and dynamic effects design basis," insofar as it requires that structures, systems, and components (SSCs) important to safety be designed to accommodate the effects of environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, and that these SSCs be protected from dynamic effects (e.g., the effects of missiles, pipe whipping, and discharging fluids) that may result from equipment failures.

GDC-16, "Containment design," as it relates to the reactor containment establishing an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

GDC-19, "Control room," insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent, to any part of the body, for the duration of the accident.

GDC-41, "Containment atmosphere cleanup," insofar as it requires systems to (1) control fission products, hydrogen, oxygen and other substances which may be released into the reactor containment; (2) reduce the concentration and quality of fission products released to the environment following postulated accidents; and (3) control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

Additionally, the regulation at 10 CFR 50.44, "Combustible gas control for nuclear power reactors," provides standards for combustible gas control in light-water-cooled power reactors.

4.1.1 Short Term Temperature and Pressure Response

LTR NEDC-33006P states that operation in the MELLLA+ range may change the break energy for the design-basis accident (DBA) recirculation suction line break (RSLB). This may impact the short term containment response. Because of this, a plant-specific evaluation is necessary to determine whether the peak drywell pressure and temperature increase. The NRC staff finds this to be acceptable since it ensures that the safety analysis will be performed and that it will be appropriate for every plant proposing to use MELLLA+.

4.1.2 Containment Dynamic Loads

LTR NEDC-33006P states that the results of the short term containment response evaluation are used to evaluate the impact of MELLLA+ on the LOCA containment dynamic loads. Since the short term temperature and pressure response is plant-specific, the determination of containment dynamic loads is also plant-specific. The NRC staff finds this to be acceptable since it ensures that the safety analysis will be performed and that it will be appropriate for every plant proposing to use MELLLA+.

4.1.3 Containment Isolation

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]] then the LTR states that a containment isolation systems evaluation

will be performed and reported in the plant-specific MELLLA+ submittal. The NRC staff finds this to be acceptable since it ensures that the safety analysis will be performed and that it will be appropriate for every plant proposing to use MELLLA+.

4.1.4 GL 89-10

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]] then the LTR states that an evaluation of the GL 89-10 program will be performed and reported in the plant-specific M+SAR. The NRC staff finds this to be acceptable since it ensures that the safety analysis will be performed and that it will be appropriate for every plant proposing to use MELLLA+.

4.1.5 GL 89-16

GL 89-16, "Installation of a Hardened Wetwell Vent," requested installation of a hardened wetwell vent system. One of the design requirements of the hardened vent system is the ability to exhaust energy equivalent to 1 percent of the current licensed thermal power. [[

]]therefore, a revised hardened vent analysis is not required.

4.1.6 GL 95-07

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]] then an evaluation of the GL 95-07 program will be performed and reported in the plant-specific M+SAR. The NRC staff finds this to be acceptable since it ensures that the safety analysis will be performed and that it will be appropriate for every plant proposing to use MELLLA+.

4.1.7 GL 96-06

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]] an evaluation of the GL 96-06 program will be performed. The NRC staff finds this to be acceptable since it ensures that the safety analysis will be performed, if necessary, and that it will be appropriate for every plant proposing to use MELLLA+.

4.2 ECCS

This section discusses the MELLLA+ impact on the following topics:

1. high pressure coolant injection (HPCI) system

2. high pressure core spray (HPCS)
3. core spray (CS) or low pressure core spray (LPCS)
4. low pressure coolant injection (LPCI) system
5. automatic depressurization system (ADS)
6. ECCS NPSH

LOCAs are postulated accidents that would result in the loss of reactor coolant from piping breaks in the RCPB at a rate in excess of the capability of the normal reactor coolant makeup system to replenish it. Loss of significant quantities of reactor coolant would prevent heat removal from the reactor core, unless the water is replenished. The reactor protection and ECCSs are provided to mitigate these accidents. The NRC staff's review covered: (1) the licensee's determination of break locations and break sizes; (2) postulated initial conditions; (3) the sequence of events; (4) the analytical model used for analyses and calculations of the reactor power, pressure, flow, and temperature transients; (5) calculations of PCT, total oxidation of the cladding, total hydrogen generation, changes in core geometry, and long-term cooling; (6) functional and operational characteristics of the reactor protection and ECCS systems; and (7) operator actions. The NRC's acceptance criteria are based on: (1) 10 CFR 50.46, insofar as it establishes standards for the calculation of ECCS performance and acceptance criteria for that calculated performance, and (2) 10 CFR Part 50, Appendix K, insofar as it establishes required and acceptable features of evaluation models for heat removal by the ECCS after the blowdown phase of a LOCA; (3) draft GDC-40 and -42, insofar as they require that protection be provided for ESFs against the dynamic effects that might result from plant equipment failures, as well as the effects of a LOCA; and (4) draft GDC-37, -41, and -44, insofar as they require that a system to provide abundant emergency core cooling be provided so that fuel and clad damage that would interfere with the emergency core cooling function will be prevented. Specific review criteria are contained in SRP Sections 6.3 and 15.6.5 and other guidance provided in Matrix 8 Table of RS-001.

4.2.1 HPCI System

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 system is to maintain reactor vessel coolant inventory in the event of a small break LOCA that does not immediately depressurize the reactor vessel. In this event, the HPCI system maintains reactor water level and helps depressurize the reactor vessel. 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.

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The system design injection rate must be sufficient for compliance with the system limiting criteria, such as meeting the ATWS vessel pressure requirements, to maintain the reactor water level above TAF at the MELLLA+ conditions. The system performance must be confirmed in the plant-specific application.

Sufficient NPSH must be available for the HPCI pump for projected operation at MELLLA+. Systems using the suppression pool as the makeup source may potentially lead to cavitation concerns following a LOCA or an ATWS event from MELLLA+ conditions. System performance relative to potential changes in the makeup source conditions must be addressed in the plant-specific application.

4.2.2 HPCS System

The HPCS system, used in BWR/5 and 6 plants, is designed to spray water into the reactor vessel over a wide range of operating pressures. 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. In this event, the HPCS system maintains reactor water level and helps depressurize the reactor vessel. This system also provides spray cooling for long-term core cooling after a LOCA. In addition, 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.

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The system design injection rate must be sufficient for compliance with the system limiting criteria, such as meeting the ATWS vessel pressure requirements, to maintain the reactor water level above TAF at the MELLLA+ conditions. The system performance must be confirmed in the plant-specific application.

Sufficient NPSH must be available for the HPCS pump for projected operation at MELLLA+. Systems using the suppression pool as the makeup source may potentially lead to cavitation concerns following a LOCA or an ATWS event from MELLLA+ conditions. System performance relative to potential changes in the makeup source conditions must be addressed in the plant-specific application. The plant-specific HPCI evaluation must also include the impact of the high suppression pool temperature on the HPCI pump temperature limit.

4.2.3 CS or LPCS

The CS/LPCS system is automatically initiated in the event of a LOCA. When operating in conjunction with other ECCSs, the CS/LPCS system is required to provide adequate core cooling for all LOCA events. There is no anticipated change in the reactor pressures at which the CS/LPCS is required. The primary purpose of the CS system is to provide reactor vessel coolant inventory makeup for a large break LOCA and for any small break LOCA after the reactor vessel has depressurized. It also provides long-term core cooling in the event of a LOCA.

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The plant-specific application will include CS/LPCS system capability to perform its functions during accidents and non-accident events. The M+SAR will provide evaluation of the systems ability to meet the NPSH design limit and any equipment temperature limit.

4.2.4 LPCI System

The LPCI mode of the residual heat removal (RHR) system is automatically initiated in the event of a LOCA. The primary purpose of the LPCI mode is to provide reactor coolant makeup for a large break LOCA and for any small break LOCA after the reactor vessel has depressurized.

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The plant-specific application will include LPCI system capability to perform its functions, during accidents and non-accident events. The M+SAR will provide evaluation of the systems ability to meet the NPSH design limit and any equipment temperature limit.

4.2.5 ADS

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 systems to inject coolant into the reactor vessel.

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4.2.6 ECCS NPSH

Since the MELLLA+ operating range does not result in an increase in heat addition to the suppression pool following a LOCA, Station Blackout (SBO), and Appendix R fire, [[

Therefore, the ECCS net positive suction head values following a LOCA, SBO, or Appendix R fire event would remain bounded by the current evaluation.

MELLLA+ may increase the heat addition to the suppression pool following a limiting ATWS. The NPSH performance for applicable systems during a postulated ATWS must be confirmed in the plant-specific application. The NRC staff's evaluation of the determination of suppression pool temperature for the anticipated transient without scram (ATWS) event is provided in Section 9.3.1.2 of this SE.

4.2.7 Conclusion

The licensee's plant-specific submittal will confirm the acceptability of the ECCS performance consistent with the surveillance test results and projected MELLLA+ conditions. Therefore, the approach described in LTR NEDC-33006P is acceptable to the NRC staff.

4.3 ECCS PERFORMANCE

The regulation at 10 CFR 50.46 delineates the acceptance criteria for the ECCS-LOCA analysis as follows:

1. The peak fuel cladding temperature should not exceed 2200 °F;

2. The total oxidation shall not exceed 0.17 times the total cladding thickness before oxidation;

3. The total local amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all the metal in the cladding cylinder surrounding the fuel, excluding the cladding surrounding the plenum volume were to react;

4. The core geometry shall be such that the core remains amenable to cooling; and

5. After successful initial operation of the ECCSs, the calculated core temperature shall be maintained at an acceptable low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

The ECCSs are designed to provide protection against postulated ECCS-LOCA 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 Appendix K to 10 CFR Part 50.

As shown in Table 4-2 of this SE, the LOCA PCT calculation will be evaluated [[ ]] This section: (1) discusses the impact of MELLLA+ operation on the ECCS-LOCA response; (2) presents sensitivity ECCS-LOCA analyses performed at the MELLLA+ statepoints; (3) specifies the ECCS-LOCA analyses that will be provided in the plant-specific MELLLA+ applications; and (4) covers the basis for the [[ ]] of the non-PCT ECCS-LOCA acceptance criteria.

Table 4-2 Disposition of ECCS-LOCA Items

Topic	MELLLA+ Effect	Disposition
Large Break Peak Clad Temperature	Small Effect	[[
Small Break Peak Clad Temperature	Negligible Effect	
Local Cladding Oxidation	Negligible Effect	
Core Wide Metal Water Reaction	Negligible Effect	
Coolable Geometry	None	
Long-Term Cooling	None	
Flow Mismatch Limits	None	]]

4.3.1 Large Break LOCA PCT

The DBA large break LOCA is performed at the EPU rated conditions, during the implementation of EPU. The maximum power level will not change for operation at the MELLLA+ domain.

However, the MELLLA+ reduced CF or high bundle power/flow initial condition significantly affects the large break ECCS-LOCA response. The MELLLA+ large break ECCS-LOCA response is similar to the ECCS-LOCA response for operation at the extended load line limit analysis (ELLLA) and MELLLA low-flow regions. The MELLLA minimum CF ECCS-LOCA response data shows that the large break LOCA PCT calculated at the reduced CF statepoint results in a more limiting PCT relative to the DBA-LOCA PCT calculated at rated conditions.

The MELLLA minimum CF statepoint corresponds to OLTP at 75 percent CF in comparisons to 120 percent OLTP at 80 percent CF for MELLLA+ minimum flow statepoint. Therefore, similar response is expected for operation at MELLLA+ reduced flow conditions. In addition, the proposed MELLLA+ slope extends from the minimum CF statepoint at the 120 percent power to the 55 percent CF statepoint. The ECCS-LOCA analysis would be performed at the limiting statepoints through out the licensed domain within the MELLLA+ upper boundary and the maximum CF.

4.3.1.1 Impact of Reduced Flow

LTR NEDC-33006P contains discussion of the impact of the reduced CF on the DBA-LOCA PCT response. The PCT response for large break LOCA has two peaks. The first PCT response occurs early in the event during the CF coast down and is determined by BT. The second peak occurs during the core uncovery and reflooding stage.

Reduced bundle power/flow condition causes BT to occur earlier and potentially lower within the bundle. In addition, the reduced CF increases the initial subcooling in the down-comer water inventory so that the mass flow through the break is greater in the early phase of the LOCA event. The BT that occurs before the jet pump uncovery is referred to as early BT. MELLLA+ has two effects on the BT and the first peak PCT. Similar to the ELLLA and MELLLA low-flow PCT response, the MELLLA+ low-flow ECCS-LOCA response results in early BT that may penetrate lower in the fuel bundle as the CF is reduced. However, the impact of the earlier BT on the LOCA PCT depends on the plant-specific conditions and response.

The LTR NEDC-33006P states [[

]] Therefore, if the lower bundle flow has a small effect on the first peak PCT, then there is little effect of the first peak on the second peak. Note that the licensing basis PCT is usually determined by the second peak PCT, even at the MELLLA+ low CFs.

4.3.1.2 Sensitivity Analysis

LTR NEDC-33006P provides generic ECCS-LOCA analyses for typical BWR/3, BWR/4, and BWR/6 plants over the MELLLA+ operating domain. The generic analyses were performed at the MELLLA and the MELLLA+ domain statepoints, using both nominal and 10 CFR Part 50, Appendix K, assumptions.

Table 4-6 and Table 4-7 of this SE show the sensitivity of large break DBA-LOCA to power level and low-flow changes for a representative BWR/3, 4, and 6. The analyses were performed at MELLLA line (100P/100F, 100P/80F) and EPU/MELLLA+ (120 percent P/100 percent CF, 120 percent P/80 percent CF, 100 percent P/55 percent CF). Table 4-6 shows that the 100 percent P/55 percent CF MELLLA+ statepoint is generally more limiting.

The expectation is that the 120 percent P/80 percent CF minimum CF statepoint should have higher PCT than rated EPU and MELLLA statepoints; however, the results in Table 4-6 show lower PCT values for the 120 percent P/80 percent CF MELLLA+ statepoint than for the 100 percent P/80 percent CF MELLLA statepoint on some cases. This behavior is even more pronounced in Table 4-7, where there is no apparent phenomenological trend under which condition the highest PCT will occur.

A number of factors contribute to the limiting PCT value including: flow redistribution, early BT due to higher bundle power-to-flow ratio, plant-specific ECCS parameters, and the initial operating limits assumed. The inconsistent results in Table 4-6 and Table 4-7are due to: (1) application of off-rated limits at MELLLA+ statepoints, and (2) different conservative assumptions used in the calculations of the ECCS-LOCA analysis for some of the plants.

Since ECCS-LOCA analysis is not performed on cycle-specific basis, the SAFER/GESTR LOCA analysis is evaluated for the MELLLA low-flow core condition, using the same ECCS inputs as the rated condition (Reference 52). In addition, if sufficient margin is available, additional conservatisms are assumed in the analysis so as to preclude re-analysis due changes in the cycle-specific plant response.

For the MELLLA+ operation, the 55 percent low-flow statepoint is most limiting for the analyses performed. However, [[

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This assumption is based on the fact that in order to meet the higher off-rated OLMCPR, the plant will operate with lower bundle powers, possibly through changes in the inserted control rod inventory. The assumption that the hot bundles are operating at lower thermal limits reduces the impact of MELLLA+ on the PCT at the 55 percent CF statepoint. GE response to RAI 25b (Reference 29) discusses their proposed approach for determining the statepoint in which the off-rated limit will be applied.

The inconsistent credit for off-rated multipliers partially account for the differences observed in Table 4-6 and Table 4-7 between MELLLA and MELLLA+ minimum CF calculation. Table 4-8 also shows large break LOCA results for Plant F performed at different power flow conditions.

4.3.1.3 Plant-Specific Analysis

LTR NEDC-33006P states that "The plant-specific M+SAR will include calculations of the Appendix K and Nominal PCT at rated power/rated CF, rated power/MELLLA+ boundary and the low-flow point on the MELLLA plus boundary at which off-rated thermal limits begin to apply (versus the 55 percent CF point). [[

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It is important to note that the LOCA analysis must be consistent with plant operating conditions. For example, the LOCA analysis at the 80 percent CF statepoint must be performed, using rated thermal operating limits, not off-rated, as would be the case in the core monitoring system. GE's response to RAI 25b (Reference 29) provides additional clarification to the proposed DBA-LOCA approach stating:

1. The MELLLA+ plant submittals will include calculations for the Appendix K and Nominal PCT at rated EPU power/rated CF, rated EPU power/minimum CF and at a low-flow point on the MELLLA+ boundary, at which the off-rated flow dependent LHGR or MAPLHGR setdown begins to apply.

2. This point will be at or above 55 percent CF on the MELLLA+ boundary between the 55 percent and the minimum CF statepoints, hitherto referred to as transition statepoint. The

ECCS-LOCA analysis at minimum CF and the transition statepoints will be initialized at the rated power LHGR and MAPLHGR limits. However, initial MCPR for the transition statepoint will apply power dependent MCPR multiplier to the assumed rated power MCPR.

3. Since credit is taken for these off-rated limits, the plants will be required to apply these limits, during core monitoring.

4. The Licensing Basis PCT, considering all calculated statepoints as described, will be reported in the plant-specific M+SAR.

The NRC staff finds this approach acceptable, with the following limitation.

ECCS-LOCA Off-rated Multipliers Limitation

a) The plant-specific application will provide the 10 CFR Part 50, Appendix K, and the nominal PCTs calculated at the rated EPU power/rated CF, rated EPU power/minimum CF, at the low-flow MELLLA+ boundary (Transition Statepoint). For the limiting statepoint, both the upper bound and the licensing PCT will be reported. The M+SAR will justify why the transition statepoint ECCS-LOCA response bounds the 55 percent CF statepoint. The M+SAR will provide discussion on what power/flow combination scoping calculations were performed to identify the limiting statepoints in terms of DBA-LOCA PCT response for the operation within the MELLLA+ boundary. The M+ SAR will justify that the upper bound and licensing basis PCT provided is in fact the limiting PCT considering uncertainty applications to the non-limiting statepoints.

b) LOCA analysis is not performed on cycle-specific basis; therefore, the thermal limits applied in the M+SAR LOCA analysis for the 55 percent CF MELLLA+ statepoint and/or the transition statepoint must be either bounding or consistent with cycle-specific off-rated limits. The COLR and the SRLR will contain confirmation that the off-rated limits assumed in the ECCS-LOCA analyses bound the cycle-specific off-rated limits calculated for the MELLLA+ operation. Every future cycle reload shall confirm that the cycle-specific off-rated thermal limits applied at the 55 percent CF and/or the transition statepoints are consistent with those assumed in the plant-specific ECCS-LOCA analyses.

c) Off-rated limits will not be applied to the minimum CF statepoint.

d) If credit is taken for these off-rated limits, the plant will be required to apply these limits during core monitoring.

4.3.1.4 ECCS-LOCA Axial Power Distribution Evaluation

Considering the assumed axial power profiles in the SAFER/GESTR methodology, LTR NEDC-33173P (Reference 37) cites the conclusions from recent sensitivity analysis. [[

]] Since for large break LOCA, the core uncovery and reflooding occur rapidly, the impact of top and mid-peaked power profile on the duration of the hot node uncovery is not as significant as its impact on the small break. For the small break, the upper nodes experience uncovery earlier and reflood later. Recent sensitivity analyses based on the current fuel design show that the top-peaked power shapes can result in a higher PCT for small breaks than comparable calculations, assuming a mid-peaked axial shape, given that the nodes higher in the core remain uncovered longer. [[

]] However, for large break LOCA, the mid-peaked power shape was found to result in more limiting PCT. GE states that large break LOCA usually results in more limiting PCTs.

In terms of axial power distribution, the NRC staff concludes that for small break LOCA, the SAFER/GESTR LOCA analysis should include the mid and top-peaked power distribution for application involving implementation of maximum operating domains. This conclusion is based on the review of EPU applications, which indicate that small break LOCA PCT does increase with EPUs. In addition, the large break ECCS-LOCA PCT is expected to be higher for operation at the minimum CF conditions at EPU power levels, characteristic of the operation at the higher operating domain.

The NRC staff confirmatory calculations indicate that the difference in PCT could be up to 200 °F when top-peaked power shape results are compared to mid-peaked power shape results. Plant-specific analyses performed by GE show a difference of approximately 150 °F. In these specific applications, even a modest increase in PCT could have a significant impact in the plant's ability to meet the ECCS-LOCA PCT requirements. Therefore, the best alternative approach to resource intensive plant-specific PCT margin evaluation is to amend the SAFER/GESTR licensing methodology.

ECCS-LOCA Axial Power Shape Limitation

For MELLLA+ applications, the small and large break ECCS-LOCA analyses will include top-peaked and mid-peaked power shape in establishing the MAPLHGR and determining the PCT. This limitation is applicable to both the licensing bases PCT and the upper bound PCT. The plant-specific applications will report the limiting small and large break licensing basis and upper bound PCTs.

4.3.1.5 ECCS-LOCA PCT Reporting

Although it may not be the limiting ECCS-LOCA PCT value, only the rated ECCS-LOCA response was being reported in the SRLR, the COLR, the regulatory reporting documents, and the applications. In GE's response to RAI 25b (Reference 29), GHNE agreed to change future SAFER/GESTR analyses and SRLRs as follows:

1. The SAFER/GESTR report will provide the Licensing Basis PCT considering all calculated statepoints. The Licensing Basis PCT will be calculated either using the previous Licensing Basis PCT plant variable uncertainty (e.g., NEDE-23875-1-PA, Section 3.1.3) or with a plant variable uncertainty specific to the calculated statepoint with the highest Appendix K PCT. Only one Licensing Basis PCT will be reported because it is the single PCT, which considers all required licensing conservatism.

2. Only SRLRs, for both MELLLA+ plants and non-MELLLA+ plants, which report these future SAFER/GESTR analyses will report the Licensing Basis PCT considering all calculated statepoints as described above. No change will be made in SRLR reporting of previous SAFER/GESTR analyses."

3. Section 6 of NEDC-32950P [Reference 53] will be revised to include determining the Licensing Basis PCT considering all calculated statepoints as described above.

4. The Initial MCPR assumed in the ECCS/LOCA analyses is reported in the SRLR.

The RAI response limits the reporting to the licensing bases PCTs and appears to exclude the Upper Bound PCT calculations, which use conservative models. In addition, the statement "a plant variable uncertainty specific to the calculated statepoint with the highest Appendix K PCT," in Item 1 is not clear. The discussion and proposed approach in Item 1 seems to differ from the current SAFER/GESTR methodology of incorporating the plant configuration uncertainties.

Therefore, the M+SAR will discuss the differences between the currently licensed approach and the proposed modified approach for NRC staff review and approval. The NRC staff will determine on plant-specific bases, whether reporting one licensing bases PCT is acceptable. Alternatively, GHNE can supplement LTR NEDC-33006P and provide the supporting information and justification for further review and approval. The latter approach can resolve the proposed methodology generically.

The NRC staff reviewed the proposed reporting approach and concludes that: (1) both the nominal and Appendix K PCTs should be reported for all of the calculated statepoints; and (2) the plant-variable and uncertainties currently applied will be used, unless the NRC staff specifically approves different plant variable uncertainty methods for application to the non-rated statepoints. Items 1, 2, and 3 of RAI 25b response (Reference 29) address the reporting of the limiting ECCS-LOCA PCT response calculated at different statepoints. The approach provided in Items 1, 2, and 3 are acceptable, with the following limitation:

ECCS-LOCA Reporting Limitation

a) Both the nominal and Appendix K PCTs should be reported for all of the calculated statepoints, and

b) The plant-variable and uncertainties currently applied will be used, unless the NRC staff specifically approves a different plant variable uncertainty method for application to the non-rated statepoints.

4.3.1.6 Conclusion

The ECCS-LOCA analysis will be performed on plant-specific bases, as discussed above. The NRC staff reviewed the content of the LTR, the sensitivity analyses, the responses to the RAIs, and concludes that the proposed approach is acceptable given the limitations discussed above.

4.3.2 Small Break LOCA PCT

LTR NEDC-33006P (Reference 1) and response to RAI 26 (Reference 30) discuss the impact of operation at the MELLLA+ conditions on the small break LOCA response.

4.3.2.1 Impact of Power Level on Small Break LOCA

EPU power level affects the small break LOCA PCT significantly because the ADS blowdown time increases due to: (a) the higher initial steam flow and (b) the increased decay heat levels. This leads to a later ECCS injection time; therefore, the time during which the fuel in uncovered increases, leading to higher PCT values. Plant-specific analysis and NRC staff confirmatory analysis have demonstrated that small break LOCA becomes the limiting LOCA at EPU conditions (120 percent P/99 percent CF).

For plants that operate with ELTR1/2, the base full break spectra are performed at the EPU power levels. For plants that uprated based on the CPPU, limited small break LOCA analysis are performed to establish the break size that produces the limiting PCT.

4.3.2.2 Impact of MELLLA+ Operation in Small Break LOCA Response

For MELLLA+ operation, the EPU power level does not change. Therefore, significant changes in the ECCS-LOCA response due to changes in the power levels are not expected. Revision 2

of the LTR NEDC-33006P states:[[

]]. While the effect of MELLLA+ is expected to be negligible, the MELLLA+ plant-specific SAR will include calculations for the limiting small break at rated power/rated CF and rated power/MELLLA+ boundary, if the small break PCT at rated power/rated CF is within [[ ]] of the limiting Appendix K PCT."

4.3.2.3 GHNE Assessment

In the revised RAI 25b response (Reference 29), GHNE assesses the impact of the MELLLA+ reduced flow on small break LOCA.

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GHNE states that to assure that potentially limiting breaks are analyzed at reduced flow, small breaks that are limiting or within [[ ]] of the limiting large break are analyzed at the reduced flow conditions. In general, statepoints that result in lower power levels are not included in the ECCS-LOCA analysis, because decrease in the power will reduce the PCT much more than any flow reduction.

The RAI response provided analyses for plants in which small break is limiting or within [[ ]] of the limiting large break. Table 4-3 shows the PCT results.

Table 4-3 Small Break LOCA PCT

Plant Type	ΔPCT (100 % CF) – (low CF)
BWR/4 (Loop Selection Logic)	0°F (low = 85 percent = MELLLA+)
BWR/4 (LPCI Modification)	+4°F (low = 85 percent = MELLLA+
BWR/5	-8°F (low = 80 percent = MELLLA)

GHNE states that these results show that the effects of reduced CF at the same core power are much lower than the [[ ]] screening criteria that will be used to perform the small break

LOCA for the minimum CF statepoint. RAI 25b (Reference 29) also contains the following small break LOCA results.

Table 4-4 DBA Limited LOCA PCT

Power (% OLTP)	Flow (% Rated)	DBA PCT (˚F)	Small Break PCT (˚F)
[[			

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Table 4-5 Small Break LOCA Limited

Power (% OLTP)	Flow (% Rated)	DBA PCT (˚F)	Small Break PCT (˚F)
[[			

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4.3.2.4 Staff Assessment

The NRC staff finds that there are additional competing effects that may affect the conclusions, including the limiting size and location of the small break LOCA.

When operating in the MELLLA+ extended domain (e.g., 120 percent P/80 percent CF), the core average void fraction is larger than for the previously analyzed EPU conditions (120 percent P/99 percent CF); therefore, the liquid coolant inventory in the vessel is smaller, and the fuel uncovery time will occur earlier. Furthermore, the vessel steam inventory is larger, and the blowdown time will be longer; thus, the ECCS initiation will be delayed. GHNE states that [[

]] However, the increased liquid density increases the vessel inventory in terms of mass given the same volume, so those two effects will tend to cancel each other. In addition, the change in downcomer enthalpy and the associated density change is fairly small (~6 BTU/lb).

The small break LOCA results provided do show PCT difference of less than between small break LOCA performed at rated and minimum flow MELLLA+ statepoint. The differences between the DBA and the small break LOCA are also less than [[ ]] for small break limited Plant B. However, the results presented in GHNE's response to RAI 25b (Reference 29) do not indicate if the reported PCTs are based on Appendix K, the licensing basis PCT or are nominal. The [[ ]] screening criteria are acceptable if the plant has sufficient margins to the PCT limit of 2200˚F. However, for those plants that are LOCA limited, a PCT difference of 20 ˚F can

make the difference. Therefore, the margins available need to be included in the screening criteria.

Small Break LOCA Limitation

Small break LOCA analysis will be performed at the MELLLA+ minimum CF and the transition statepoints for those plants that: (1) are small break LOCA limited based on small break LOCA analysis performed at the rated EPU conditions; or (2) have margins of less than or equal to [[ ]] relative to the Appendix K or the licensing basis PCT.

4.3.2.5 Small Break LOCA PCT Conclusion

The NRC staff concludes that small break LOCA analysis will be performed for the MELLLA+ minimum CF statepoint for those plants that: (1) are small break LOCA limited based on analysis performed at rated EPU conditions; or (2) have margins of less than or equal to [[ ]] relative to the upper bound or the licensing basis PCT. For all other plants, the NRC staff accepts GHNE's proposed [[ ]] screening criteria.

Based on the approach proposed on the assessment of the impact of MELLLA+ operation small break LOCA, the content of RAI 26 (Reference 30), the evaluation of the sensitivity analysis, and the limitation applied, the NRC staff concludes that the impact of small break LOCA for operation at the MELLLA+ domain will be accounted for in the plant-specific application.

4.3.3 Break Spectrum Shape

In revised RAI 25b (Reference 29), GHNE assessed the break spectrum shape for the new operating strategies. For BWR jet pump plants, GHNE states that the break spectrum, characterized by PCT versus break area, has always maintained a standard shape. The break spectrum shape was reconfirmed for different plant types as a result of NFI or for EPU (ELTR1/2). Note that full break spectrum analyses are performed for power uprates based on ELTR1/2. For NFI, GHNE analyzes the 80 percent and 60 percent DBA breaks. GE reports that several plants have analyzed the breaks between 60 percent and the small breaks (i.e., the full break spectrum) and in all cases the shape has not changed.

For large break LOCA, the limiting break is the maximum large break. For the small break, the limiting break size needs to be determined. For the standard break spectrum shape, there is a peak temperature at the maximum break size. The peak temperature decreases with break size, because of the lower inventory loss through the break. The peak temperature decrease with break size trend continues until the lower break flow is no longer sufficient to depressurize the reactor system.

For these break sizes, the limiting single failure is the failure that results with loss of the high pressure ECCS systems and would require the ADS to depressurize the reactor system in a timely manner so that the low pressure ECCS can inject. In this small break range, a PCT occurs during the inventory loss through the break during the reactor system depressurization phase, before the low pressure ECCS injection occurs.

The NRC staff accepts that in overall the break spectrum shape may not change significantly for the new operating strategies, including EPU and MELLLA+. However, it is feasible that the limiting break size may change slightly for plant-specific application and it would also depend on the methods employed by specific code. However, the same SAFER/GESTR is currently employed.

For CPPU applications, only three small break sizes are analyzed, in which case the PCT size is determined in 0.01 square inch interval. Therefore, if there is shift in the limiting break size in the spectrum, the PCT small break LOCA break size may not be captured. Since small break LOCA is becoming limiting for EPU and consequently MELLLA+ operation, performing sufficient small break LOCA analyses to establish the limiting break that yields the highest PCT is important.

Break Spectrum Limitation

The scope of small break LOCA analysis for MELLLA+ operation relies upon the EPU small break LOCA analysis results. Therefore, the NRC staff concludes that for plants that will implement MELLLA+, sufficient small break sizes should be analyzed at the rated EPU power level to ensure that the peak PCT break size is identified.

4.3.4 Single Failures

GHNE Assessment

For the large break case, several single failures combination of the available ECCS are evaluated. In general, the limiting single failure is the one that causes the least amount of ECCS flow and results in later core reflooding times. Although typically one single failure scenario is dominant, some plants have two failures, with almost the same large break PCT.

For the small break, the single failure that causes the loss of all of the high pressure ECCS makeup and the largest number of low-pressure ECCS is always the limiting case. If the high pressure ECCS is available, core uncovery is unlikely. The second worst single failure for a small break is the ADS failure.

The NRC staff agrees with GHNE's assessment.

4.3.5 Break Location

The review of Plant F ECCS-LOCA analysis for EPU application shows that the small break LOCA became the limiting break at EPU power levels (Reference 52). In addition, the SAFER/GESTR small break LOCA analysis performed at current licensed thermal power (CLTP) and EPU power levels resulted in changes in both the limiting break (DBA to small break LOCA) and the location of the break (from recirculation suction to discharge). This raised the concern that ECCS-LOCA response at EPU power levels can result in changes in the break locations as the available ECCS-LOCA network changes with the limiting break changes. Since plants that uprated with CLTR will not perform full break spectrum as was the case for Plant F, break location changes will not be determined.

In the revised RAI 25b response (Reference 29), GHNE provided assurances that break location change would occur for plants that implemented specific modification to their LPCI system actuation. In general, the recirculation line break is always the limiting break location, because it has the largest piping and is located low on the reactor vessel.

Large Break LOCA

For all plants, the recirculation suction line is the limiting large break. For specific BWR/4 plants that implemented LPCI modification, discharge line break are also considered, because LPCI will flow out of the discharge break after the LPCI suction side is isolated from the LPCI injection location. In this configuration, LPCI (into the broken loop) would flow out of a discharge break

but not out of a suction break. The large recirculation line discharge break is considered but is not limiting.

Small Break LOCA

For all plants, except BWR/4 plants with the LPCI modification, the recirculation suction line is the limiting small break. For BWR/4 plants with the LPCI modification, the recirculation discharge break is limiting, because LPCI (into the broken loop) would flow out of a discharge break but not out of a suction break. Plant F is a LPCI modification plant, therefore, the limiting break location is expected to switch from suction to discharge if the limiting break size changes from large break to small break.

NRC Staff Assessment

As discussed above, the change in the break location is attributed to specific modification, which is known to result in change in the break location, if the small break LOCA becomes more limiting. Therefore, the change in the break location for Plant F would not have been missed if full break spectrum analysis is not performed as is the case for CPPU plants. Therefore, the NRC staff finds GHNE's explanation provided in the revised RAI 25b acceptable.

4.3.6 10 CFR 50.46 Acceptance Criteria

The PCT change due to MELLLA+ will be calculated on a plant-specific basis for the limiting large break LOCA to demonstrate compliance with the 2200 °F acceptance criterion of 10 CFR 50.46. The PCT affects cladding oxidation. Higher PCT values at the MELLLA+ reduced flow conditions will affect the amount of cladding oxidation. However, as long as the PCT remains below 2200 °F, the local oxidation and core-wide metal-water reaction acceptance criteria of 10 CFR 50.46 are met. For plants with low margin to the PCT and non-jet-pump plants, the M+SAR will provide confirmation that they meet the acceptance criteria of 10 CFR 50.46.

4.3.7 Recirculation Drive Flow Mismatch Limits

Limits have been placed on recirculation drive flow mismatch over a range of CF rates. For most plants, the limits on flow mismatch are more relaxed at lower CF rates. The drive flow mismatch affects the CF coastdown following the break, because one of the recirculation pumps is operating at lower speed and will therefore coast down faster, due to lower stored inertial energy. The lower flows associated with MELLLA+ have a significant effect on the recirculation pump coast down, which impacts the ECCS-LOCA analysis results. This impact will be included in the required ECCS-LOCA plant-specific calculations. [[

]]

4.3.8 Conclusion

The NRC staff evaluated the impact of MELLLA+ operation on the: (1) large break LOCA, (2) small break LOCA; (3) break spectrum; (4) single failure; and (4) break location. Plants-specific applications will provide large break LOCA PCT analysis for the MELLLA+ statepoints. The M+SAR will also include small break LOCA if the [[ ]] screening criteria is met. Changes in the break location will be analyzed for those plants, in which the LPCI modification was implemented. Based on the discussion provided in Section 4.3, the RAI responses, and

the limitations applied, the NRC staff finds that plant-specific MELLLA+ applications will account for the impact of the MELLLA+ operation on the ECCS-LOCA response.

4.4 MAIN CONTROL ROOM ATMOSPHERE CONTROL SYSTEM

There is no impact as there is no change in the source terms or the release rates.

4.5 STANDBY GAS TREATMENT SYSTEM

There is no impact as the primary and secondary containment leak rates do not change
[[
]]

4.6 MSIV LEAKAGE CONTROL SYSTEM

Many BWR licensees have removed the MSIV leakage control systems. LTR NEDC-33006P states that a plant-specific evaluation will be provided for those plants that have the system. The NRC staff finds this to be acceptable since it ensures that the safety analysis will be performed and that it will be appropriate for every plant proposing to use MELLLA+.

4.7 POST-LOCA COMBUSTIBLE GAS CONTROL SYSTEM

There is no change in [[
]] therefore, there is no change in the production of hydrogen and oxygen and, therefore, MELLLA+ has no effect on the post-LOCA combustible gas control system.

Table 4-6 Typical LOCA Analysis Results for MELLLA+

Power/Flow Point[1]		100P/100F (Rated)	100P/80F (MELLLA)	120P/100F (EPU)	120P/80F (MELLLA+)	100P/55F (MELLLA+)
		PCT[2]				
Plant Type		[[				
BWR/3	Nominal					
	Appendix K					
BWR/4	Nominal					
	Appendix K					
BWR/6	Nominal					
	Appendix K					]]

(1) Power level shown is percent of OLTP

(2) [[
]]

Table 4-7 Plant-Specific LOCA Analysis Results for MELLLA+

Power/Flow Point[1]		Rated 105P/100F	EPU 120P/100F	MELLLA 105P/85F	MELLLA+ 120P/85F	MELLLA+ 100P/68F
PCT[2]						
Plant		1st / 2nd Peak, °F	1st / 2nd Peak, °F	1st / 2nd Peak, °F	1st / 2nd Peak, °F	1st / 2nd Peak, °F
218 BWR/4	Power/Flow	[[				
	Nominal					
	Appendix K					
251 BWR/4	Power/Flow					
	Nominal					
	Appendix K					
218 BWR/6	Power/Flow					
	Nominal					
	Appendix K					]]

(1) Power level shown is percent of OLTP

(2) [[

]]

Table 4-8 Reduced Core Flow MELLLA & MELLLA+ (PLANT F Data; Large BWR/4)

REGION	CORE POWER	CORE FLOW (% rated)	LOCA Analysis Type	GE13 °F	GE14 °F
MELLLA	[[				
MELLLA					
MELLLA+					
MELLLA+					
MELLLA+					
MELLLA+					
Rated (EPU)					
Rated (EPU)					]]

(1) All cases analyzed the DBA recirculation suction line break with battery failure

(2) All PCTs are 2nd peak limited

5.0 INSTRUMENTATION AND CONTROL

This section provides the NRC staff review of the instrumentation and control. Table 5-1 lists the specific topics and the corresponding NRC staff dispositions for plant-specific application.

Table 5-1 Instrumentation and Control Topics

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

Plant-specific evaluations will be reported in the plant-specific submittal consistent with the applicable limitations. The applicability of the generic assessments for a plant-specific application will be evaluated and the plant-specific submittal will either document the confirmation of the generic assessment or provide a plant-specific evaluation if the generic applicability assessment does not apply.

5.1 NSSS MONITORING AND CONTROL

LTR NEDC-33006P disposition of the principal NSSS monitoring and control topics are provided in Table 5-2 below. Changes in the process parameters resulting from the MELLLA+ operating range expansion and their effects on instrument performance and setpoints are evaluated in the following sections.

Section 11.1, addresses the TS changes associated with the instrument allowable values and setpoints. Section 5.3 of this SE covers the effect of the MELLLA+ operation on the neutron monitoring system (NMS) instrumentation setpoints.

Table 5-2 Disposition of NSSS Monitoring and Control Topics

Topic	MELLLA+ Effect	Disposition
APRMs, Intermediate Range Monitors (IRMs), and Source Range Monitors (SRMs)	Minimum, except for bypass voiding impact	[[
LPRMs	Minimum, except for bypass voiding impact	
Rod Block Monitor (RBM)	None	
Rod Worth Minimizer (RWM)/ Rod Control and Information System (RCIS)	None	]]

Since the maximum power does not increase for implementation of expanded operating domain operation, the effects on the performance of the NMS are limited. The LTR NEDC-33006P states that the following evaluations of the NMS are applicable to GE or Reuter Stokes supplied monitoring equipment, or other equipment that meets GE specifications.

The functions of the NMSs are as follows:

The source range monitor (SRM) system provides neutron flux information during reactor startup and is used to monitor the core during fuel loading and refueling operations.

The IRM system provides neutron flux information during startup and heat-up operation. The IRMs generate trip signal to mitigate conditions that can result in local fuel damage. There are eight (8) IRM detectors in the BWR/3 and BWR/6 cores. Some BWR/4 cores have six (6) IRM detectors.

The LPRM system provides signals proportional to the local neutron flux from different locations in the reactor core. The signals generated by the individual LPRM elements provide fuel cladding protection and reactor core performance monitoring by combining by the various NMSs (e.g., APRMS, RBM, etc.) to initiate scrams, rod blocks or core power monitoring,.

The LPRM detector signals at different locations and elevations are combined in the APRMs. The function of the APRMs are to: (1) detect core-wide neutron flux transients and generate trip signal that generates automatic reactor scram before the reactor experiences conditions outside the safety and licensing design basis; (2) block control rod withdrawals if the reactor settles outside the licensed power/flow domain; and (3) provide an indication of the core average power level for operation in the power range.

The TIP detectors operate in a guide tube in each LPRM string assembly. The primary function of the TIP detectors is to calibrate the LPRMs. However, TIP readings are also used in the core simulator systems for monitoring of the fuel operating conditions, assessment of the fuel thermal limits margin, and evaluation of the core neutronic and thermal-hydraulic performance.

The RBM system initiates control rod withdrawal block that prevents exceeding the SLMCPR during withdrawal of single control rod. The RBM system also provides indication to the operator of the change in the relative local power during control rod withdrawal movements.

The following sections evaluate the impact of operation in the expanded operating domain on the performance and reliability of the NMSs.

5.1.1 Assessment

5.1.1.1 APRMs, IRMs, and SRMs

The LTR NEDC-33006P [[ ]] the effect of MELLLA+ operation on the APRMs. During the EPU implementation, the APRM output signals are calibrated to read 100 percent at the CLTP. [[

]] Using normal plant surveillance procedures, the IRMs may be adjusted to ensure adequate overlap with the SRMs and APRMs. The NRC staff agrees with APRM assessment. Section 5.1.1.5 discusses the impact of bypass voiding on the APRMs.

5.1.1.2 LPRMs

There is no change in the neutron flux experienced by the LPRMs and TIPs, resulting from the MELLLA+ operating range expansion. Therefore, [[

]] Section 5.1.1.5 of this SE discusses impact of bypass voiding on the LPRMs.

5.1.1.3 RBM

The RBM uses LPRM instrumentation inputs that are combined and referenced to an APRM channel. The LTR NEDC-33006P states that [[

]] The NRC staff concurs with this assessment. Section 5.1.1.5 of this SE discusses impact of bypass voiding on the RBM.

5.1.1.4 Rod Worth Minimizer (RWM) and Rod Control and Information System (RCIS)

The LTR NEDC-33006P states that the RWM and RCIS are normal operating systems that do not perform a safety related function. The RWM and RCIS rod pattern controller functions 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. The region in which the RWM and RCIS are active is unaffected by MELLLA+ operation. The NRC staff agrees with the LTR NEDC-33006P assessment that [[ ]]

However, the NRC staff also finds that enforcing rod patterns for power levels below 30 percent and limiting excessive control rod withdrawal after a higher power level is reached do serve as a safety function, because the control rod drop accident consequence is minimized at low power and the SAFDLs are protected in the event of a reactivity initiated event. Therefore, although the associated safety analyses may not take credit for initiation of these NMSs, there is an associated safety function.

5.1.1.5 Additional Review Topics

Depending on control cell loading in terms of the number of high powered bundles with exposure, the operation in the MELLLA+ high power/low-flow conditions will result in non-solid bypass condition that affects the accuracy and reliability of the NMSs. LTR NEDC-33173P addressed the impact of bypass voiding on the reliability and effectiveness of the NMS, during steady state and transient conditions. The main conclusions are summarized below.

5.1.1.5.1 Steady State Bypass Voiding

The detector design specifications for the NMS (e.g., LPRMs) limits the bypass voiding to 5 percent. For EPU and MELLLA+ operation, the bypass voiding could be 5 percent or higher at the exit. NEDC-33173P (Reference 37) contains a limitation that the bypass voiding will be limited to 5 percent for the LPRM D-level for implementation of EPU and MELLLA+. The steady state bypass voiding will be reported in the SRLR for every reload.

The instrumentation specification design basis limits the presence of bypass voiding to 5 percent (LRPM levels). Limiting the bypass voiding to less than 5 percent for long-term steady operation ensures that instrumentation is operated within the specification. For EPU and MELLLA+ operation, the bypass voiding will be evaluated on a cycle-specific basis to confirm that the void fraction remains below 5 percent at all LPRM levels when operating at steady-state

conditions within the MELLLA+ upper boundary. The highest calculated bypass voiding at any LPRM level will be provided with the plant-specific SRLR.

5.1.1.5.2 Stability Setpoint Setdown

During some transients, such as RPT events, the hot channel bypass voiding could reach a maximum of 32 percent, depending on the code used. The high in-channel and bypass voids will primarily affect the LPRM detectors by reducing the detector response, assuming the same power in the adjacent fuel bundle. This reduction in detector response is due to a decrease in the moderation caused by the presence of high in-channel and bypass voids in the upper part of the fuel bundle. The in-channel and bypass voids decrease the thermal neutron flux incident on the detectors for the same neutron flux generated in the adjacent fuel. Table 1-3 provides the bypass voiding at various operating domains (MELLLA+ and MELLLA), using different codes. The NRC staff concludes that the instrument calibration error is less than 5 percent for OPRM cells and less than 2 percent for APRM signals. There is a limitation that requires setdown of the instrumentation to preclude the presence of the high in-channel and bypass voiding for EPU and MELLLA+ conditions. The specific setdown value is dependent upon the stability solution employed.

5.1.1.5.3 Steady State Thermal TIP Readings Above the LPRM D-level

The LTR NEDC-33173P (Reference 37) limitation restricts the bypass voiding at the LPRM D-level to 5 percent during steady state operation. If, for operation at the high power/low-flow MELLLA+ 55 percent CF statepoint, the bypass voiding above the LRPM D-level is higher than the 5 percent specification limit, then there could be an impact on the thermal TIPs affecting the calibration of the LPRMs, and also on the core simulator axial power distribution adaption.

Plants are not expected to operate the reactor at the 55 percent CF statepoints, where the power level is around the OLTP. However, operators need to be cognizant of the fact that sustained operation at the 55 percent CF statepoint due to EOOS or during plant maneuvers may affect the TIP readings (especially thermal TIPs) and the core simulator adaption feature. Statepoint where the bypass voiding greater than 5 percent could occur above the D-level is plant- and cycle-specific and needs to be identified and justified. Therefore, the following limitation applies for operation at the MELLLA+ domain.

<u>Bypass Voiding Above the D-level Limitation</u>

Plant-specific MELLLA+ applications shall identify where in the MELLLA+ upper boundary the bypass voiding greater than 5 percent will occur above the D-level. The licensee shall provide in the plant-specific submittal the operator actions and procedures that will mitigate the impact of the bypass voiding on the TIPs and the core simulator used to monitor the fuel performance. The plant-specific submittal shall also provide discussion on what impact the bypass voiding greater than 5 percent will have on the NMS as defined in Section 5.1.1.5. The NRC staff will evaluate on plant-specific bases acceptability of bypass voiding above D level.

Impact of Bypass Voiding Greater than 5 Percent and Power Distribution Uncertainties

One component of the power distribution uncertainties, σ_{P4b} is derived from the thermal TIP measured/calculated comparisons. With the presence of bypass voiding as could potentially occur for the 55 percent CF statepoint or along the MELLLA+ boundary, the reliability of the TIP measurement (specially thermal TIPs) will be affected. Therefore, for the top part of the fuel bundle above the LPRM D-level, the 4-bundle uncertainty at the lower flow statepoints cannot be quantified based on the thermal TIP reading and core tracking data. In addition, the gamma

scan benchmarking data will not represent operation at the lower flow statepoints earlier in the cycle. The gamma scan benchmarking data will be representative of end of cycle conditions, which are not in the MELLLA+ region. However, the data will characterize the cumulative effects of higher power (EPU) and/or lower flow statepoints (MELLLA+) earlier in the cycle as they affect isotope production and any effects on the core monitoring instrumentation. However, the sensitivity to early and mid cycle operating conditions has not been evaluated. Therefore, the power distribution uncertainties applied at the MELLLA+ boundary between the 80 percent and the 50 percent CF statepoints where the voids will be highest cannot be validated by gamma scan data, specifically for the different uncertainty components (e.g., σ-peak and σ-bundle) as applied to the SLMCPR. The same power distribution uncertainties applied at the rated conditions are applied at the reduced CF statepoints, and GHNE did not propose an alternative approach.

The SLMCPR calculation shows sensitivity to CF and FWF uncertainties; the CF and FWF uncertainties increase with decreasing CF and FWF (See Figure 2-6 of this SE). The higher CF and FWF uncertainties must be applied to the non-rated conditions, and should be high enough to compensate for the difficulties associated with benchmarking the reduced CF conditions. Therefore, the NRC staff concludes that for the 55 percent CF statepoint and along the MELLLA+ upper boundary up to the minimum CF statepoint, the highest reduced CF uncertainty will be applied. This is consistent with the CF uncertainty applied to the SLO operation. The NRC staff finds with the increased CF uncertainty applied consistently at the MELLLA+ upper boundary statepoints, the application of the rated power distribution uncertainty for the specific components are acceptable. However, this does not exclude confirmation that σ_{P4b} is applicable where bypass voiding above the D-level is not present, such as the minimum CF statepoint. In addition, the NRC staff assessment is based on the SLMCPR calculational methodology in which the base thermal-hydraulic condition at the minimum CF and the 55 percent CF statepoints are determined and perturbed according to the associated uncertainty components. Section 2.2.1.1, "SLMCPR," of this SE also contains discussion on the CF uncertainties.

5.1.2 Conclusion

The plant-specific application will provide confirmation of the impact of bypass voiding on the reliability of the NMSs as discussed above. Based on the conditions noted and the assessment covered in this section, the NRC staff accepts the adequacy of the NMS for operation at MELLLA+ condition.

5.2 BOP MONTIORING AND CONTROL

Section 5.2 of LTR NEDC-33006P discusses the BOP monitoring and control systems. The instruments that monitor and the controls that directly interact with or control reactor parameters are usually within the NSSS. The other monitoring and control instrumentation are defined as BOP. The topics covered in the BOP monitoring and control instrumentation in Section 5.2 of the LTR are as follows:

1. Pressure Control System

2. Turbine Steam Bypass System

3. Feedwater Control System

4. Leak Detection System

For MELLLA+, GE evaluated the BOP systems and determined that these systems can be [[ ]] GE has determined that in general MELLLA+ does not affect the system except for the setpoint change for APRM flow biased scram, which is evaluated below. As stated in the LTR, the plant-specific submittal will confirm the [[

]] will be addressed in the plant-specific submittal. Any major changes to BOP monitoring and control are addressed in the plant-specific MELLLA+ submittal, therefore, the NRC staff finds the proposed approach acceptable.

5.2.1 Regulatory Evaluation

The NRC staff has used the following regulatory basis for its evaluation of Section 5.2:

1. 10 CFR 50.36(c)(1)(ii)(A)

The regulation at Paragraph (c)(1)(ii)(A) of 10 CFR 50.36, Technical specifications, states, in part, that where a limiting safety system setting (LSSS) is specified for a variable on which a safety limit (SL) has been placed, the setting must be so chosen that an automatic protective action will correct the abnormal situation before a safety limit is exceeded. The analytical limit (AL) is the limit on the process variable at which the instrument loop protective action occurs as assumed in the plant's safety analysis. Protective action at the AL ensures that the SL is not exceeded. The AL, however, does not account for uncertainties associated with the instrument loop. The instrument loop uncertainty is accounted for during calculation of an instrument loop's trip setpoint (TSP). Accordingly, limits for instrument channels that initiate protective functions must be included in the TSs. Setpoints found to exceed TS limits are considered a malfunction of an automatic safety system. Such an occurrence could challenge the integrity of the reactor core, reactor coolant pressure boundary, containment, and associated safety systems.

1. Regulatory Guide 1.105, "Setpoints for Safety-Related Instrumentation"

RG 1.105 is used to endorse Part 1 of ISA-S67.04-1994 and describes a method acceptable to the staff for complying with NRC's regulations for ensuring that setpoints for safety-related instrumentations are initially within and remains within the technical specification limits. The RG lists four exceptions to the standard in regard to crafting an acceptable setpoint methodology. The two exceptions which were taken into consideration for this license amendment were that the LSSS is being specified as a technical-specification-defined limit in order to satisfy the requirements of 10 CFR 50.36 (Exception # 3) and that the allowable value's relationship to the setpoint methodology and testing requirements in the TSs must be documented (Exception # 4). In addition, the NRC issued a Regulatory Issue Summary 2006-17 on August 24, 2006, which provided NRC staff position on the requirements of 10 CFR 50.36, "Technical Specifications," regarding limiting safety system settings during periodic testing and calibration of instrument channels.

5.2.2 Instrument Setpoint Methodology Evaluation

GE stated that the determination of allowable values (AV) and setpoints include consideration of measurement uncertainties. The setpoints and AVs are derived from the analytical limits (AL) used in specific licensing or SEs. The settings are selected with sufficient margin to minimize

inadvertent initiation of the protective action, while assuring that adequate margin is maintained between the system settings and the actual limits. GE has indicated that they want to use simplified process to determine the instrument AV and setpoint for MELLLA+ applications. The NRC staff has previously reviewed the simplified approach and had accepted in the review of LTR NEDC-33004P (Reference 35) with certain conditions. The NRC staff asked GE to confirm if these conditions will be met for NEDC-33006P. GE in its RAI response (Reference 10) has reiterated these conditions, which are as follows:

1. No pressure increase

2. NRC approved GE or plant-specific methodology

   [[

   ]]

Based on the GE's RAI response in Reference 10, the NRC staff has determined that the instrument setpoint based on the NRC-approved methodology will meet the requirements of 10 CFR 50.36(c)(1)(ii)(A) and the guidance in RG 1.105 and is therefore acceptable to the staff.

5.3 TECHNICAL SPECIFICATION INSTRUMENT SETPOINTS

GE has identified only two instrument setpoints, which may be affected by the LTR NEDC-33006P, which are identified below:

5.3.1 APRM Flow-Biased Scram

The MELLLA+ APRM flow biased scram AL line is established [[

]] GE has used the simplified approach as discussed before to calculate the AV. MELLLA+ does not apply to SLO, so the SLO setpoints are unchanged. This TS change is classified as plant-specific and will be reviewed by the NRC staff for licensees that seek NRC approval to adopt this LTR. Since this setpoint is calculated based on the NRC-approved setpoint methodology and the analysis for the MELLLA+ implementation which have been reviewed by the NRC staff as discussed above, the NRC staff finds the proposed change acceptable.

5.3.2 Rod Block Monitor

The RBM setpoints are established to mitigate RWE during power operation. GE has determined that [[

]] because RWE event is evaluated for each reload and any cycle specific adjustments will be performed per the COLR. Based on this the NRC staff find this approach acceptable.

6.0 ELECTRICAL POWER AND AUXILIARY SYSTEMS

This section provides the NRC staff review of the electrical power and auxiliary systems. Table 6-1 lists the specific topics and the corresponding NRC staff dispositions for plant-specific application.

Table 6-1 Electrical Power and Auxiliary Systems Topics

Section	Title	[[	
6.1	AC Power		
6.2	DC Power		
6.3	Fuel Pool		
6.4	Water Systems		
6.5	Standby Liquid Control Systems (SLCS)		
6.7	Fire Protection		]]

For the topics dispositioned generically, the plant-specific submittal will confirm and document the applicability of the generic assessments or provide a plant-specific evaluation.

6.1 AC POWER

The NRC staff reviewed Section 6.1 of LTR NEDC-33006P to verify the GE's contention that there will be no changes to the parameters that would impact AC power requirements. Based on this review, the NRC staff agrees with GE's assessment. Therefore, the NRC staff concludes that the implementation of MELLLA+ is acceptable.

6.2 DC POWER

The NRC staff reviewed Section 6.2 of LTR NEDC-33006P to verify the GE's contention that there will be no changes to the parameters that would impact DC power requirements. Based on this review, the NRC staff agrees with GE's assessment. Therefore, the NRC staff concludes that the implementation of MELLLA+ is acceptable.

6.3 FUEL POOL

The NRC staff reviewed Section 6.3 of LTR NEDC-33006P to verify the GE's contention that there will be no changes to the parameters that would impact the operation of the fuel pool due to the expansion of the CF operating range. Based on this review, the NRC staff agrees that the fuel pool will not be affected by the proposed expansion of the power/flow map [[ ]] Therefore, the NRC staff concludes that the implementation of MELLLA+ is acceptable for the fuel pool.

6.4 WATER SYSTEMS

The NRC staff reviewed Section 6.4 of LTR NEDC-33006P to verify the GE's contention that there will be no changes to the parameters that would impact the operation of the water systems due to the expansion of the CF operating range. Based on this review, the NRC staff agrees that the water systems will not be affected by the proposed expansion of the power/flow map as none of the parameters, that affect the water systems (i.e.,[[ ]]) will be changed. Therefore, the NRC staff concludes that the implementation of MELLLA+ is acceptable for the water systems.

6.5 STANDBY LIQUID CONTROL SYSTEM (SLCS)

The SLCS provides backup capability for reactivity control independent of the control rod system. The SLCS functions by injecting a boron solution into the reactor to affect shutdown. The LTR NEDC-33006P reviewed the impact of MELLLA+ operation on the functional capability of the system to deliver the required amount of boron solution into the reactor.

The NRC's acceptance criteria are based on (1) GDC-27 and -28, insofar as they require that at least two independent reactivity control systems, preferably of different design principles, be provided, with both systems capable of making and holding the core subcritical from any hot standby or hot operating condition, sufficiently fast to prevent exceeding acceptable fuel damage limits; (2) GDC 29, insofar as it requires that at least one of the reactivity control systems be capable of making the core subcritical under any condition sufficiently fast to prevent exceeding acceptable fuel damage limits; and (3) 10 CFR 50.62(4), insofar as it requires that the SLCS be capable of reliably injecting a borated water solution into the reactor pressure vessel at a boron concentration, boron enrichment, and flow rate that provides a set level of reactivity control. Specific review criteria are contained in SRP Section 9.3.5 and other guidance provided in Matrix 8 Table of RS-001.

The SLCS is typically a manually operated system, but a few BWRs have automatic actuation. The topics addressed in the LTR evaluation are provided in Table 6-2.

Table 6-2 Reactivity Control Topics

Topic	MELLLA+ Effect	Disposition
SLCS Shutdown Margin	Potential increase in boron requirements. Reflected in cycle-specific SRLR	[[
Strong Rod Out (SRO) Shutdown Margin	Shutdown margin may change. Reflected in cycle-specific SRLR	
Hot shutdown boron weight	Potential increase in boron requirements. Reflected in cycle-specific EOPs	
System hardware	Potential increase in reactor pressure for system operation	
ATWS requirements	Potential increase in the boron injection rate requirements	]]

6.5.1 Assessment

6.5.1.1 SLCS Cold and Hot Shutdown Boron Weight

[[

]] The LTR NEDC-33006P states that 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 B10 in the stored neutron absorber solution. The implementation will be plant-specific and documented in the plant-specific M+SAR.

6.5.1.2 System Hardware

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. [[

]] The effect, if any, of the increased vessel pressure on SLCS performance will be incorporated in the plant-specific ATWS analyses.

6.5.1.3 ATWS Requirements

The ATWS analysis for MELLLA+ operating range conditions (Section 9.3.1 of this SE) may impose new boron injection rate requirements. The LTR NEDC-33006P states that 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 B^{10} in the stored neutron absorber solution. An evaluation of the plant-specific ATWS requirements will be provided as part of the plant-specific M+SAR and will be incorporated in the plant-specific ATWS analyses.

6.5.2 Conclusion

The licensee's plant-specific submittal will confirm the acceptability of the system performance consistent with the surveillance test results and projected MELLLA+ conditions. Therefore, the approach described in LTR NEDC-33006P is acceptable to the NRC staff.

6.6 HEATING, VENTILATING, AND AIR CONDITIONING

There is no impact on the systems in the turbine building, reactor building, and the drywell, which support normal plant operation as the process temperatures and heat load from motors and cables do not change. Therefore, the NRC staff finds the implementation of MELLLA+ acceptable.

6.7 FIRE PROTECTION

The NRC staff reviewed Section 6.7 of LTR NEDC-33006P to verify the GE's contention that there will be no changes to the parameters that would impact the operation of fire protection due to the expansion of the CF operating range. Based on this review, the NRC staff agrees that [[

]]. Therefore, the NRC staff concludes that the implementation of MELLLA+ is acceptable for fire protection.

6.8 OTHER SYSTEMS AFFECTED

Those systems that are significantly affected by the MELLLA+ operating range expansion are addressed by the LTR. Any other systems not addressed by the LTR are not significantly affected by the MELLLA+ operating range expansion.

7.0 POWER CONVERSION SYSTEMS

This section addresses the evaluations in Chapter 10 of Regulatory Guide (RG) 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," (Revision 3 in three parts, ADAMS Accession Nos. ML011340072, ML011340108, and ML011340116)," that are documented in the CLTR. The MELLLA+ core operating range expansion does not affect the power conversion systems. The pressure, steam and feedwater flow rates, and fluid temperature ranges do not change.

7.1 TURBINE GENERATOR

The NRC staff reviewed Section 7.1 of LTR NEDC-33006P to verify the GE's contention that there will be no changes to the parameters that would impact the operation of the turbine generator due to the expansion of the CF operating range. Based on this review, the NRC staff agrees that the turbine generator will not be affected by the proposed expansion of the power/flow map because none of the parameters that affect the turbine generator (i.e., steam pressure, steam flow, or electrical output) will be changed. Therefore, the NRC staff concludes that the implementation of MELLLA+ is acceptable for the turbine generator.

7.2 CONDENSER AND STEAM JET AIR EJECTORS

The NRC staff reviewed Section 7.2 of LTR NEDC-33006P to verify the GE's contention that there will be no changes to the parameters that would impact the operation of the condenser and steam jet air ejectors due to the expansion of the CF operating range. Based on this review, the NRC staff agrees that the condenser and steam jet air ejectors will not be affected by the proposed expansion of the power/flow map, because none of the parameters which affect the condenser and steam jet air ejectors (i.e., steam pressure or flow rate) will be changed. Therefore, the NRC staff concludes that the implementation of MELLLA+ is acceptable for the condenser and steam jet air ejectors.

7.3 TURBINE STEAM BYPASS

The NRC staff reviewed Section 7.3 of LTR NEDC-33006P to verify the GE's contention that there will be no changes to the parameters that would impact the operation of the turbine steam bypass due to the expansion of the CF operating range. Based on this review, the NRC staff agrees that the turbine steam bypass will not be affected by the proposed expansion of the power/flow map, because none of the parameters which affect the turbine steam bypass (i.e., steam pressure or flow rate) will be changed. Therefore, the NRC staff concludes that the implementation of MELLLA+ is acceptable for the turbine steam bypass.

7.4 FEEDWATER AND CONDESNATE SYSTEMS

The NRC staff reviewed Section 7.4 of LTR NEDC-33006P to verify the GE's contention that there will be no changes to the parameters that would impact the operation of the feedwater and condensate systems due to the expansion of the CF operating range. Based on this review, the NRC staff agrees that the feedwater and condensate systems will not be affected by the proposed expansion of the power/flow map, because none of the parameters which affect the feedwater and condensate systems (i.e., FWT, pressure, or flow rate) will be changed. Therefore, the NRC staff concludes that the implementation of MELLLA+ is acceptable for the feedwater and condensate systems.

8.0 RADWASTE SYSTEMS AND RADIATION SOURCES

The radioactive waste and radiation protection areas have been previously reviewed for the MELLLA operating range report. This evaluation looks only at the difference from the MELLLA to the MELLLA+ operating range. The NRC's acceptance criteria for radioactive waste systems and radiation sources are based on GDC-60, "Control of releases of radioactive materials to the environment," -61, "fuel storage and handling and radioactivity control," and -64, "monitoring radioactivity releases," the regulation at 10 CFR 50.34a, "Design objectives for equipment to control releases of radioactive material in effluents - nuclear power reactors," and the design objectives specified in Appendix I to 10 CFR Part 50.

8.1 LIQUID AND SOLID WASTE MANAGEMENT

LTR NEDC 33006P indicates that the power level, feedwater flow, and steam flow do not change for the MELLLA+ operating range expansion, therefore, the volume of liquid radwaste and the coolant concentrations of fission and corrosion products will be unchanged. The volume of waste generated is not expected to increase [[

]] The LTR also indicates that coolant fission and corrosion product levels will be evaluated on a plant-specific basis. The NRC staff finds the LTR evaluation to be acceptable.

8.2 GASEOUS WASTE MANAGEMENT

Radiological releases from gaseous effluents are administratively controlled to remain within existing limits. Gaseous releases are affected by fuel cladding performance, main condenser air inleakage, charcoal adsorber inlet dew point, and charcoal adsorber temperature. [[

]] The NRC staff finds the LTR evaluation to be acceptable.

8.3 RADIATION SOURCES IN THE REACTOR CORE

During power operation, the radiation sources in the core are directly related to the fission rate, while post-operation the radiation sources result from accumulated fission products. [[

]] The NRC staff finds the LTR evaluation to be acceptable.

8.4 RADIATION SOURCES IN REACTOR COOLANT

Activation, activation corrosion, and fission products make up the radiation sources in the reactor coolant.

For coolant activation products, the short-lived radionuclide nitrogen (N)-16 is one of the primary contributors to the radiation dose in the turbines during operation. Since the neutron flux and steam flow will not change with the MELLLA+ operating range expansion, there should be no change in the coolant activation products.

Fission products are in the steam component and reactor water. The activity in the steam consists of noble gases from the core plus carryover activity from the reactor water. The fission product activity in the steam and reactor water is the result of fission products escaping from the fuel rods. Since the core power level and fuel thermal limits are not changed with the MELLLA+ operating range expansion, the releases from the fuel should not change. [[

]]

Activated corrosion products are the result of metallic materials in the reactor water being activated in the core region. The feedwater flow, steam flow, and power do not change with the MELLLA+ operating range. [[

]]

The NRC staff finds the LTR evaluation of radiation sources in the reactor coolant to be acceptable. The LTR indicates that plant-specific evaluations must be performed to evaluate whether there is potential [[ ]] resulting in higher levels of fission products in the steam.

8.5 RADIATION LEVELS

Plant radiation levels for normal and post-shutdown operation are related to the radionuclide inventory in the reactor coolant (steam and water) except where the core is directly involved. Under MELLLA+, the radionuclide concentrations should not vary significantly, because the power or flow rate do not change; therefore, radiation dose rates in the plant should not change. [[

]] The LTR indicates that normal operational, post-shutdown, and post-accident radiation levels are to be addressed on a plant-specific basis. The LTR should indicate that a

plant-specific evaluation should be performed to evaluate the radiation dose rates in post-accident sampling locations. The NRC staff finds the LTR evaluation of radiation levels to be acceptable.

8.6 NORMAL OPERATION OFF-SITE DOSES

During normal operations, airborne releases from the offgas system and gamma shine from the plant turbines are the primary sources of the off-site radiation dose. There is no change in the core power and the steam flow rate. [[

]] The NRC staff finds the LTR evaluation of normal operation off-site doses to be acceptable.

9.0 REACTOR SAFETY PERFORMANCE EVALUATIONS

This section provides the NRC staff review of the reactor safety performance evaluations. Table 9-1 lists the specific topics and the corresponding NRC staff dispositions for plant-specific application.

Table 9-1 Reactor Safety Performance Evaluation Topics

Section	Title	[[	
9.1	AOOs		
9.2	DBA		
9.3	Special Events		]]

Plant-specific evaluations will be included in the plant-specific submittal consistent with the format and level of detail as discussed in LTR NEDC-33006P 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.

9.1 AOOS

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, -17, and -20.

GDC-10 requires that the reactor core and associated control and instrumentation systems be designed with sufficient margin to ensure that the SAFDLs 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 RCPB are not exceeded during normal operating conditions and AOOs.

GDC-17 requires that an onsite electric power system and an offsite electric power system shall be available to provide sufficient capacity and capability to assure that SAFDLs and design conditions of the RCPB are not exceeded as a result of AOOs.

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

The SRP provides the following:

1. 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;"

2. 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;

3. An incident of moderate frequency should not generate a more serious plant condition unless other faults occur independently; and

4. 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-specific update final safety analysis report (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.

9.1.1 AOO Assessment

9.1.1.1 Fuel Thermal Margin Events

The limiting transient analyses are performed on cycle- and core-configuration-specific bases during the standard reload analyses. The plant's limiting transient analyses are specified in the plant-specific UFSAR. The analyses are performed according to the NRC-approved GHNE licensing methodology GESTAR II (Reference 36). The GHNE licensing methodology identifies the following transients as typically the most limiting events that set the OLMCPR:

1. Generator Load Rejection without Bypass (LRNBP)

2. Turbine Trip without Bypass Failure (TTNBP),

3. Feedwater Controller Failure (FWCF) – Maximum Demand,

4. Loss of Feedwater Heating (LFWH) or Inadvertent HPCI Startup,

5. Pressure Regulator Downscale Failure (BWR/6 Only), and

6. Control Rod Withdrawal Error.

Transients performed during the reload are not limited to the above listed transients. Fuel Loading Errors (FLE) are also evaluated as an AOO, during reload analysis in accordance with GESTAR II licensing methodology. GESTAR Amendment 28 was recently approved which re-categorized the FLE as an accident and may no longer be considered an AOO. The NRC staff will follow up on this issue on plant-specific basis. Any transient analysis identified as limiting in the plant-specific UFSAR should also be included in the reload analysis set. Additional transients such the single recirculation pump seizure event are also analyzed during NFI.

The LTR NEDC-33006P provided transient analyses performed at the MELLLA+ minimum flow statepoint in order to establish if the event category or response will change with operation at the MELLLA+ operating domain. Table 9-2 below presents LRNBP, TTNBP, FWCF and LFWH response for a BWR/4 and a BWR/6 plants. These two plants are referred to as Plant D and Plant E in LTR NEDC-33173P.

The results in Table 9-2 provides the event results initiated from the 120 percent power at ICF conditions and 120 percent power at 85 percent CF (MELLLA+) statepoints. The LTR NEDC-33006P states that [[

]] Data provided in Tables 9-3 and 9-4 were obtained from the audit documents show the same transient response for two plants, but includes the transient response initiated from the 55 percent CF MELLLA+ statepoint.

Comparisons of the responses from Plants D and E show that the transient response initiated from the 55 percent CF statepoint differ for the two plants, with the highest response for Plant E (BWR/6) occurring at the 55 percent CF statepoint. In this case, the 55 percent MELLLA+ statepoint response bounds the ICF response. To limit the higher ΔCPR response for operation at the off-rated MELLLA+ domain, licensees will apply the off-rated limits. Application of these off-rated multipliers will require operation at lower bundle powers and peak fuel nodal powers possibly through changes in the inserted control rod inventory. Whether plants can operate the higher bundle at the low-flow MELLLA+ boundary and meet the off-rated limits will be demonstrated on plant-specific bases. Section 9.1.1.3 of this SE discusses the power- and flow-dependent limits.

For both cases, [[ ]] Most EPU plants cannot achieve ICF. However if licensed the transient initiated from ICF will be determined. In addition, based on data provided in Tables 9-2, 9-3, and 9-4, the TTNBP event bounds the typically limiting FWCF pressurization transients. Therefore, plants would need to include the TTNBP in their pressurization transients.

Table 9-2 Typical AOO Event Results Summary

Event	Parameter	Units	120 % OLTP ICF	120% OLTP 85% CF
[[				

Event	Parameter	Units	120 % OLTP ICF	120% OLTP 85% CF
				]]

Table 9-3 Plant D Pressurization Transient (BWR/4, ODYN)

Transient	P/F %EPU	ΔCPR(B)	TOP %	MOP%	Generic K(P)	Calculated K(P)
[[						
						]]

Table 9-4 Plant E Pressurization Transient (BWR/6, ODYN)

Transient	P/F% EPU	Δ CPR(A)	TOP%	MOP%	Generic K(P)	Calculated K(P)
[[						
						]]

For MELLLA+ operation, licensees are expected to migrate to TRACG best-estimate code for AOOs, which provides higher CPR margin relative to ODYN. GHNE's response to RAI 13 provided the pressurization transient response for Plant D, using TRACG. Table 9-5 below shows the pressurization transients change in the CPR relative to the nominal normalized by the initial CPR (e.g., ΔCPR/initial CPR (ICPR)) response at different powers and flow conditions, including the MELLLA+ 55 percent CF knee statepoint. It is difficult to assess the CPR

response with power and flow without knowing the corresponding TRACG ICPR for each statepoint, which varies.

However, for the TRACG cases, the typically limiting pressurization transients LRNBP and FWCF do not bound the TTNBP event for all cases. The ΔCPR/ICPR response at the reduced flow conditions appears lower than the higher flow cases for the same power levels. The highest change in CPR occurs at the lower power at higher flow condition for the limiting pressurization events.

Table 9-5 Plant D Thermal Margin with Power and Flow (TRACG)

Power (% OLTP) /Core Flow (%rated)	LRNBP ΔCPR/ICPR	TTNBP ΔCPR/ICPR	FWCF ΔCPR/ICPR
[[			
			]]

Table 9-6 below provides comparisons of the LFWH event initiated near the rated CF and the minimum CF statepoint for the EPU power levels. This table provided in the audit documents shows that the LFWH event is more limiting at the 85 percent CF statepoint. In addition, it can be seen that the LFWH event can be more limiting in terms of mechanical overpower response compared to the pressurization events. This is a slow transient with increased subcooling and with corresponding power increase with no anticipatory RPT and scrams occurs when the high neutron flux setpoint is reached.

Table 9-6 Plant D LFWH Transient

Transient	P/F	ΔCPR	TOP percent	MOP percent
[[				
				]]

The transient results provided indicate that while the MELLLA+ statepoint could be limiting, or the limiting event for the limiting set of transients may change, the transient response results do not indicate any unexpected changes or severity. For MELLLA+ operation, plants will perform the pressurization transients at all the statepoints including the ICF, minimum CF, and the 55 percent CF statepoints. This is consistent with the current practice where transient analysis initiated from the minimum MELLLA statepoints (e.g., 105 percent P/82 percent CF) is calculated. The M+SAR will provide the plant-specific transient response initiated from these statepoints or supplement the MELLLA+ application with SRLR, which will show the cycle-specific MELLLA+ response at these statepoints.

The NRC staff finds the proposed AOO approach for MELLLA+ operation acceptable because:

1. the limiting pressurization response will be performed on cycle- and core-configuration-specific basis during the standard reload;

2. the analyses will be performed using NRC-approved analytical methods;

3. the analyses performed will be based on the NRC-approved licensing methodology and the plant-specific UFSAR;

4. the transients initiated from the specific MELLLA+ statepoints will be analyzed;

5. the plant-specific application will submit the SRLR, which will contain the cycle-specific limiting response, including the non-pressurization transients; and

6. the transient results provided do not indicate unexpected changes in the overall transient responses.

9.1.1.2 Rod Withdrawal Error

The rod withdrawal error (RWE) is an abnormal operational transient which affects only a limited number of fuel assemblies in the core. The local and radial peaking factors can increase substantially in the fuel assemblies in the immediate vicinity of the withdrawn control rod. Thus, this transient is of safety concern with regard to potential fuel rod overheating (i.e., MCPR) and clad overstraining (i.e., 1 percent plastic strain). [[

]] Table 9-7 provides the mean ΔCPR that corresponds to the generic RBM setpoints.

Table 9-7 ARTS RBM Setpoints

RBM Setpoint	Power/Flow	Mean ΔCPR
[[		
		]]

A similar study was performed in order to assess the adequacy of the generic RBM setpoints for operation at MELLLA+ conditions. Table 9-8 gives the results.

Table 9-8 MELLLA+ Effect on RWE ΔCPR

RBM Setpoint	Power percent (OLTP) /Flow	Mean ΔCPR
[[		
		]]

[[ ]]
However, the results show that RBM setpoint of 1.18 can result in higher and significant mean ΔCPR value. The examination of the data shows that for the proposed operating strategy, the generic RBM value may not provide equivalent [[.]] Therefore, the ARTS [[ ]] needs to be expanded with additional data from plants operating in the MELLLA+ domain. The NRC staff concludes that plants operating at the MELLLA+ expanded operating domains need to perform RWE analyses and confirm the RBM setpoints.

RWE Limitation

Plants operating at the MELLLA+ operating domain shall perform RWE analyses to confirm the adequacy of the generic RBM setpoints. The M+SAR shall provide a discussion of the analyses performed and the results.

9.1.1.3 Power- and Flow-Dependent Limits

MELLLA+ may affect the transient response from limiting off-rated statepoints. Table 9-9 below provides a sample set of ICPR values different operating conditions, including the 55 percent CF, 93 percent OLTP conditions in the MELLLA+ domain. The TASC ICPR is set by iteration such that the transient MCPR is equal to the MCPR safety limit. The PANACEA ICPR is a nominal prediction from the PANACEA 3D Simulator. For LRNBP, TTNBP, and FWCF the PANACEA ICPR is from the nuclear state used as an input to the ODYN 1D transient calculation.

Since the TASC ICPR (SLMCPR +)CPR) accounts for the transient change in CPR, the higher TASC ICPR would indicate a more limiting event. Based on the results provided, the most limiting event initiated from the 55 percent CF statepoint is the LRNBP for BWR/4 and TTNBP for BWR/6. In general, the ICPR determined from PANACEA, which reflects the nominal prediction, is higher (less limiting) for the lower power off-rated cases.

While this dataset provides an indication of the transient CPR response at the various power/flow conditions provided, the direct impact of MELLLA+ (change in initial CF) at off-rated power level (i.e., 93 percent) is not determined.

LTR NEDC-33006P commits to a plant-specific submittal containing the confirmation of the MELLLA+ impact on transients initiated from off-rated conditions.

The plant-specific applications will provide prediction of key parameters for cycle exposures for operation at EPU and MELLLA+. The plant-specific prediction of these key parameters will be

plotted against the EPU referenced plant experience database and MELLLA+ operating experience, if available. For evaluation of the margins available in the fuel design limits, plant-specific applications will also provide quarter core map (assuming core symmetry) showing bundle power, bundle operating LHGR, and MCPR for BOC, MOC, and EOC. Since the minimum margins to specific limits may occur at exposures other than the traditional BOC, MOC, and EOC, the data will be provided at these exposures.

Table 9-9 PANACEA / TASC ICPR Comparison

Event	Power %OLTP) / Core Flow (% Rated)	ICPR
[[		
		]]

The operating MCPR, LHGR, and/or MAPLHGR thermal limits are modified by a flow factor when the plant is operating at less than 100 percent CF. The flow dependent MCPR ($MCPR_f$) is primarily based upon an evaluation of the slow recirculation increase event. [[

]]

Similarly, the thermal limits are modified by a power factor (the power-dependent MCPR - $MCPR_p$) when the plant is operating at less than 100 percent power. This factor was generically developed for all plants and is referenced to the power level used in the reload transient analysis. [[

]] The plant-specific M+SAR will provide the confirmation of the power and flow dependent limits.

9.1.1.4 Non-Limiting Events

Table 9-1 of NEDC-33006P provides an assessment of the effect of the MELLLA+ operating range expansion for each of the Reference 36 limiting AOO events and key non-limiting events.

9.1.1.5 Additional Topics Affecting AOO

The following additional topics are discussed on the affect of MELLLA+:

9.1.1.5.1 Water Rod Modeling And Debris Filters

In the review of LTR NEDC-33006P, the NRC staff identified that some analyses did not model the water rod where the code had the modeling capability (e.g., TRACG). GHNE provided a sensitivity analysis that indicates that lumping the water rod in the bypass does not result in nonconservatism. However, GE has committed to perform future TRACG analyses using the water rod option.

Fuel bundles use debris filters, which could increase the single-phase pressure drop. The debris filters need to be included in the modeling in order to account for the additional pressure drop.

9.1.1.5.2 Fuel T-M Limits

EPU/MELLLA+ operating strategy transient response can be higher relative to the OLTP operation. The number of fuel bundles operating at the peak LHGR envelopes is expected to be higher for plants operating with 24-month cycles at EPU and MELLLA+ conditions. Therefore, the T-M overpower response during limiting AOO events can be higher for operation at EPU and MELLLA+ operating strategy. Section 3.2.6 of the SE for NEDC-33173P discusses the NRC staff review of the plant-specific licensing methodology, which ensures that plants meet the T-M overpower limit during AOOs for the fuel designs loaded in the core. The NRC staff determined that mechanical overpressure (MOP) and thermal overpressure (TOP) are calculated but not documented in the applications or the associated regulatory documents. The NRC staff concludes that the plant-specific MELLLA+ applications must include the plant's overpower response. In addition, since the transient response is cycle- and core-specific, the SRLR must report the plant T-M overpower response during the limiting transients, considering any allowed EOOS options.

In addition, the NRC staff review determined that the 40 percent depletion history assumption under the ODYN model might under predict the T-M overpower by 5 percent. Therefore the NRC staff concludes that a margin of greater than 10 percent is warranted for models unable to account for nodal void reactivity bias with exposure. The plant-specific MELLLA+ applications will provide confirmation that there is a 10 percent margin to the centerline melt and the 1 percent diametric strain acceptance criteria for the transient LHGR limit calculation.

Additionally any limitations associated with the AOO delineated in the NRC staff SE approving the most recent version of NEDC-33173P are applicable to MELLLA+.

9.1.2 AOO Conclusion

The applicability of the [[ ]] presented in LTR NEDC-33006P will be confirmed in the licensee's plant-specific submittal using an NRC-approved methodology.

The approach described in LTR NEDC-33006P is acceptable to the NRC staff with satisfactory compliance to the limitations.

9.2 DBA

GE stated that the source term is constant; therefore, [[

]] Since the DBA calculations will not be affected by the proposed expansion of the power/flow map, the NRC staff concludes that the implementation of MELLLA+ is acceptable for all of the events listed in Section 9.2 of the LTR, except for the liquid radwaste tank failure. The LTR indicates that a plant-specific evaluation of MELLLA+ impact on the liquid radwaste tank failure analysis should be performed. The NRC staff finds the LTR evaluation of DBA radiological consequences to be acceptable.

9.3 SPECIAL EVENTS

LTR NEDC-33006P considers three special events: ATWS, SBO, and ATWS with core instability. The topics addressed in the LTR evaluation are provided in Table 9-10.

Table 9-10 Special Events Topics

Topic	MELLLA+ Effect	Disposition
ATWS (Overpressure)	Less effective power reduction from RPT	[[
ATWS (Suppression Pool Temperature and Containment Pressure)	Less effective power reduction from RPT	
ATWS (PCT and Oxidation)	Insignificant change because same initial thermal margin (ICPR) and MLHGR are used for all power/flow conditions	
SBO	None	
ATWS with Core Instability	The time of initiation of divergent oscillations and the magnitude of oscillations change slightly.	]]

9.3.1 ATWS

ATWS is defined as an AOO followed by the failure of the reactor portion of the protection system specified in GDC 20. The regulation at 10 CFR 50.62 requires that:

1. each BWR have an ARI system that is designed to perform its function in a reliable manner and be independent (from the existing reactor trip system) from sensor output to the final actuation device.

2. each BWR have a SLCS with the capability of injecting into the reactor vessel a borated water solution with reactivity control at least equivalent to the control obtained by injecting 86 gpm of a 13 weight-percent sodium pentaborate decahydrate solution at the natural boron-10 (B10) isotope abundance into a 251-inch inside diameter reactor vessel.

3. each BWR have equipment to trip the reactor coolant recirculation pumps automatically under conditions indicative of an ATWS.

The NRC staff's review was conducted to ensure that:

1. the above requirements are met,

2. sufficient margin is available in the setpoint for the SLCS pump discharge relief valve such that SLCS operability is not affected by the proposed MELLLA+/EPU, and

3. operator actions specified in the plant's emergency operating procedures (EOPs) are consistent with the generic emergency procedure guidelines/severe accident guidelines (EPGs/SAGs), insofar as they apply to the plant design.

In addition, the NRC staff reviewed the MELLLA+ ATWS analysis to ensure that the following ATWS acceptance criteria are met:

1. The peak vessel bottom pressure is less than the ASME Service Level C limit of 1500 psig;

2. The PCT is within the 10 CFR 50.46 limit of 2200 °F;

3. The peak suppression pool temperature is less than the design limit; and

4. The peak containment pressure is less than the containment design pressure.

9.3.1.1 ATWS Assessment

Operation in the MELLLA+ domain affects the ATWS performance of the reactor. One of the first safety actions taken in an ATWS is a 2RPT. When operating in the MELLLA+ corner, the final power after a 2RPT is significantly higher than when operating at OLTP. This is illustrated in Figure 9-1 of this SE. This higher power following the 2RPT results in a higher integrated heat load to the containment, which affects the safety performance.

The class of ATWS events is very large (i.e., there are many ATWS scenarios). However, as a first order approximation, one could extrapolate the containment relative heat load as being proportional to the steady state power-to-flow ratio. For example, operating in the corner of MELLLA+ domain (120 percent OLTP, 80 percent CF) results in approximately 150 percent higher containment heat load than at the OLTP at rated CF.

The NRC staff agrees with the LTR NEDC-33006P, [[

]]

Because the pressure and suppression pool temperature depends on a variety of plant-specific inputs, the limiting events will be evaluated on a plant-specific basis for the M+SAR at the most limiting cycle exposure.

ATWS LOOP Limitation

As specified in LTR NEDC-33006P, [[

]] To evaluate the effect of reduced RHR capacity during LOOP, the plant-specific ATWS calculation must be performed for a sufficiently large period of time after HSBW injection is complete to guarantee that the suppression pool temperature is cooling, indicating that the RHR capacity is greater than the decay heat generation. The plant-specific application should include evaluation of the safety system performance during the long-term cooling phase, in terms of available NPSH.

The EPGs require an emergency reactor depressurization if the suppression pool temperature reaches the HCTL. HCTL is defined so that transferring all the stored energy of the pressurized primary system to the suppression pool will not result in containment integrity violation. The HCTL value is plant-specific and is a function of the operating reactor pressure. Because of the larger power-to-flow ratio when operating at the MELLLA+ corner, which results in approximately 150 percent containment higher heat load, one could expect that the HCTL will be reached in approximately 66 percent (or 100/150) of the time that it would take to reach HCTL when operating at OLTP.

BWR ATWS events tend to challenge the suppression pool temperature and peak containment pressure limits, because the SLCS is relatively slow and takes up to 2400 seconds to inject the HSBW. The MELLLA+ operation may increase the containment heat load by up to 150 percent (compared to OLTP operation at 100 percent flow); thus making the event even more challenging. An option that the NRC staff strongly encourages is to increase the boron concentration for the SLCS so that the integrated heat load to containment remains constant. For example, if the power density is increased by 10 percent, the boron injection time must be reduced by 10 percent so the integrated heat load remains constant.

TRACG simulations performed by GHNE indicate that a typical BWR operating in the MELLLA+ corner will reach the HCTL value before the reactor is shutdown by boron injection; thus, emergency depressurization will be required under MELLLA+. Previous ATWS analyses indicated that reactors operating at OLTP may or may not reach the HCTL. The new TRACG simulations indicate that the HCTL limit is reached in approximately 600 seconds, while the time required to inject the HSBW can be as high as 2400 seconds. Note that the HSBW time is very conservative, and shutdown is expected in significantly shorter times if the boron is mixed uniformly in the core; nevertheless, the calculations indicate that emergency depressurization is very likely at MELLLA+/EPU conditions, but it may or may not be required under OLTP conditions. This is a qualitative change introduced by operation in the MELLLA+/EPU domain, which affects the reactor response to ATWS events.

The NRC staff performed a number of confirmatory calculations of suppression pool temperatures during ATWS using different tools. Figure 9-2 shows a comparison of CONTAIN calculations of the suppression pool temperature based on the core response calculated by ODYN and TRACG. From this figure, it can be concluded that the ODYN results, while conservative in this case, are not necessarily bounding for all conditions. For the particular case analyzed, the ODYN-based final temperature is largest (i.e., conservative), but the time-dependent temperature calculated using TRACG is higher than the ODYN temperature for most of the transient.

Figure 9-3 shows that operation at MELLLA+ conditions result in a significantly increased pressure peak following primary system isolation, which is caused by the increase in steam flow and less effective flow coast down following 2RPT to reduce power at a higher rod line. Note that the ODYN calculation supporting this figure was performed with one SRV out of service. For this calculation, the pressure exceeds the ATWS acceptance limit in the MELLLA+ case. In a plant-specific situation, equipment that may be out-of-service must be considered in the

calculation. If the ATWS acceptance criteria are not met, the equipment must be in-service while operating in the MELLLA+ domain.

Figure 9-4 shows the suppression pool temperature calculated by ODYN/STEMP for three different reactors at conditions somewhat representative of OLTP (100P/75F) and EPU/MELLLA+ (120P/85F). Note that the rod lines depicted in this figure are not the rated rod lines, so they do not represent a valid comparison between OLTP and MELLLA+. This figure, however, illustrates the differences between different plants using a consistent model and assumptions. The different plant responses are caused by plant parameters like suppression pool volume-to-power ratio, and boron injection capability (e.g., stand-pipe injection versus HPCI system). This figure shows a very large variability of the ATWS results among plants. Therefore, the NRC staff concludes that the ATWS event for MELLLA+ cannot dispositioned generically for all BWRs.

The impact of MELLLA+ on ATWS containment performance is significant. The following sections discuss additional relevant topics, including reactor depressurization upon reaching the HCTL and NPSH availability for required equipments during ATWS.

9.3.1.2 Effect of MELLLA+ on Suppression Pool Temperature and NPSH for ECCS Equipment

9.3.1.2.1 Background

For isolation ATWS events, or events where the steam generation exceeds the capacity of the turbine bypass valves, the ultimate heat sink is the suppression pool. As steam is generated in excess of capacity, the primary cooling system pressure increases and the SRVs open, discharging steam to the suppression pool. Also, under some circumstances, a reactor depressurization is required either manually or automatically. For depressurization, the SRVs are opened and steam discharged into the suppression pool until the primary system pressure reaches a pre-defined value (typically 50 psi). Under all these scenarios the suppression pool temperature increases.

During transient events, the preferred source of cold water is the condenser; however, the condenser is not a safety source of water, and it may not be available for some events. The safety source of water for most ESFs is the suppression pool. As the suppression pool heats up, so does the cooling water available for the ESF systems like the ECCS.

The ECCS equipment has operability requirements that depend on the water temperature. Specifically, a pre-defined amount of NPSH is required to prevent cavitation of the pump propellers. The NPSH requirements determine the maximum water temperature in which a piece of ECCS equipment can operate under. NPSH requirements are determined mostly experimentally by the equipment manufacturer, and they can vary depending on the duration of the operation. For example, a small amount of cavitation can be tolerated for a short period of time without compromising the equipment integrity.

The relationship between the NPSH and temperature depends on the operating pressure (i.e., the saturation temperature at the operating pressure). If the pressure of the source of water increases (e.g., by containment over-pressurization), the ECCS equipment will be able to operate at higher water temperatures.

The NPSH requirements vary by equipment and plant, but a typical value for the HPCI system is 140 °F. If a 5-psig containment overpressure is assumed, the allowed coolant temperature can

be increased up to 170 °F for short periods of time. Low pressure and low volume injection systems have smaller NPSH requirements.

9.3.1.2.2 Suppression Pool Performance during ATWS Events

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]] Even though LOOP initiates the sequence of events with the condenser available, it soon becomes an isolation event. At this point, some emergency equipment that is available under MSIV event may not be available under LOOP. This is plant-specific. For example, in some plants, the RHR system operates at a reduced capacity without off-site power, resulting in a higher suppression pool temperature.

All BWRs define a HCTL for their suppression pool. This is a plant-specific temperature, which is defined so that the suppression pool temperature will be below containment limits after condensing all the steam required to depressurize the reactor. If the suppression pool temperature reaches the HCTL, a manual emergency depressurization is required. The rationale behind this requirement is defense-in-depth. Even though high-pressure injection may be available at this stage of the transient, it may become unavailable, which would require a depressurization to allow the use of low-pressure sources of injection. If the suppression pool temperature is above the HCTL and the reactor is at pressure, a loss of high-pressure injection becomes a fatal event, because the reactor cannot be depressurized without compromising the containment. The HCTL value is plant-specific, and it depends roughly on the ratio of suppression pool volume to the reactor power. A typical value is 160 °F.

9.3.1.2.3 Emergency Depressurization

Manual emergency depressurization is required if either the suppression pool temperature reaches the HCTL limit, or high pressure injection becomes unavailable (e.g., NPSH limits are reached because of the high suppression pool temperature). The depressurization rate depends on the SRV capacity. Typically, the depressurization phase takes approximately 5 minutes. The depressurization rate is faster when the reactor pressure is higher, and then it decreases exponentially. Figure 9-9 of this SE shows a typical pressure response during an isolation ATWS with depressurization at approximately 600 seconds into the transient, when the suppression pool temperature reached the HCTL limit.

During the depressurization stage, the pressure is continuously decreasing. This derivative on the pressure causes continuous flashing of the liquid water in the vessel as the saturation temperature decreases. This steam flashing results in a high void fraction in the core that shuts down the reactor through the negative void reactivity coefficient. Thus, during the depressurization phase the reactor is shutdown. Figure 9-5 of this SE shows the reactor power for the isolation ATWS event. As predicted, the reactor power is decreased to essentially decay heat levels when the depressurization is initiated.

During the depressurization phase, the operator is instructed to stop all sources of coolant injection into the vessel (except the SLCS, CRD and RCIC) to prevent overflowing of the vessel caused by the flashing. Thus, the NPSH and availability of ECCS equipment is not an issue during this phase. Calculations, though, predict cladding dryout during this phase. The severity of the dryout depends on the inventory of water in the vessel prior to depressurization. The ATWS management strategies with lower target-water-level result in more severe dryout.

Figure 9-6 of this SE shows the peak fuel clad temperature for three different strategies: TAF+5', TAF, and TAF-2'.

Figure 9-2 of this SE shows the suppression pool temperature calculated for the above transients. The transients were initiated from the MELLLA+ minimum CF statepoint (120 percent OLTP, 80 percent flow). The green line shows a clear inflection point at approximately 600 seconds when the depressurization starts. The red and black lines show the suppression pool temperature calculated by the licensing code ODYN with CONTAIN and STEMP pool heating codes. Both codes include the function of RHR pool cooling. ODYN does not depressurize, but it uses generally conservative assumptions. The ODYN final suppression pool temperature is larger than TRACG's, but the temperature predicted by TRACG is higher for most of the transient; it only becomes smaller when the depressurization is completed. Both codes predict very high final suppression pool temperatures (210 °F to 220 °F). The NPSH is definitely a concern for the required ECCS equipment under these conditions, including RHR.

9.3.1.2.4 Re-Criticality After Emergency Depressurization

When the depressurization phase is over, the pressure stabilizes and steam flashing stops. At this point, the reactor may become critical again and regain power if sufficient boron has not been injected. Even if the reactor becomes critical, the amount of power required to generate a critical void fraction level is significantly lower at 50 to 100 psi than at 1000 psi; therefore, after depressurization, the power level is expected to be significantly lower than before, even not accounting for the additional boron injected during the approximate 5 minute depressurization. In addition, decay heat by itself produces a significant core void fraction because the water level is maintained low enough to prevent recirculation flow, so the CF is only driven by the internal recirculation through the core bypass region. Even before sufficient boron has been injected to shutdown the reactor (i.e., HSBW), decay heat, and a reduced quantity of boron may be sufficient to maintain the reactor subcritical.

The predictions of consequences of emergency depressurization are inconclusive. Some TRACG simulations performed by GHNE indicate that, following the emergency depressurization, sufficient boron has been mixed into the core volume to maintain the reactor shutdown at the reduced pressure (approximately100 psi) by the combined effect of the mixed boron and the void fraction generated by decay heat. Other TRACG calculations performed at the request of the NRC staff showed that the reactor recovers to criticality following the depressurization (see Figure 9-5 and Figure 9-7 of this SE). For these later runs, the SRVs were forced to re-close once 50 psig is reached per EOPs. Re-closing the SRVs results in a pressure perturbation that induces re-criticality and the reactor power increases. The results of these new calculations are shown in Figure 9-5 and Figure 9-7. The re-criticality periods are apparent in Figure 9-5. They appear to be random in nature, in amplitude and duration. Most have relatively low power levels (of the order of 20 to 30 percent), but some power spikes with power greater than 100 percent are observed. Figure 9-7 shows that the reactor pressure during re-criticality periods is as high as 2 MPa (300 psi), and it has some random characteristics. For the TRACG depressurization calculations, the containment limits are satisfied; fuel suffers dryout overheat due to core uncovery (see Figure 9-6 of this SE), with a more severe transient for the lower-water-level control strategies (e.g., TAF-2 than for TAF+5 strategy).

Confirmatory TRACE calculations performed by the NRC staff show that, following the depressurization, the reactor becomes critical again and re-pressurizes. In the particular event modeled by TRACE, the reactor re-pressurizes back to almost 5 MPa (approximately 700 psi) before the pressure is reduced again when the boron concentration is sufficient to maintain the

reactor shutdown. During this period, the power level oscillates wildly, and it reaches spikes as high as 200 percent OLTP, with an average of 50 percent to 70 percent. See Figure 9-8 and Figure 9-9 of this SE. The HCTL and emergency de-pressurization start at approximately 600 seconds. Following the emergency de-pressurization, the reactor becomes critical and the pressure recovers. Following a short closure to maintain pressure above 50 psi at time approximately 1000 seconds, all SRVs are open during the re-pressurization event at time approximately 1100 seconds. The calculated suppression pool temperature indicates that containment limits would have been violated in this transient (see Figure 9-10). The HCTL is reached in ~600 seconds. The final pool temperature for the TRACE calculation is approximately 240EF.

When criticality is reached at low pressures, the TRACE calculation indicates that the SRV flow area is not sufficient to dissipate all the volumetric flow of steam produced in the core because of the lower density of low pressure steam. Therefore, the pressure increases, and it may overshoot the new equilibrium condition where steam production in the core equals the volumetric steam flow that the SRVs can accommodate at the new pressure. The overshoot may occur because of steam condensation as the pressure increases (reverse flashing effect on reactivity).

Re-criticality after depressurization is not certain. Reactor performance during the pressurization depends on plant- and event-specific parameters and/or assumed operator actions such as re-closing some SRVs after the pressure reaches the 50 psi target. The NRC staff investigated whether the BWRs operating at MELLLA+ operating domain can depressurize based on the three water level strategies, without experiencing re-criticality. Some TRACG calculations do not show re-criticality after depressurization. For example, in GHNE's response to RAI 5.1 (Reference 31), GHNE provides TRACG ATWS with depressurization calculations for all three water level strategies. Specifically, Figure 9-12 shows the reactor power decrease with depressurization until hot shutdown is reached, without re-criticality for all three water level strategies. Licensees could potentially mitigate re-criticality by employing methods to inject HSBW (e.g., earlier and faster injection like HPCS, increase the boron concentration). Figure 9-13 thru Figure 9-18 show the changes in other key core parameters as the reactor depressurizes.

As discussed earlier, the isolation ATWS analysis will be done on plant-specific bases. The NRC staff will evaluate the plant-specific ATWS performance, before approving operation at the MELLLA+ condition.

9.3.1.2.5 Containment Over-Pressure

During LOCA events, steam flows into the containment and deposits all the enthalpy directly in the containment atmosphere and structures by direct condensation. In ATWS events, the steam is directed inside the suppression pool, which absorbs most of the steam enthalpy. As the suppression pool temperature rises, there is a slow transfer of enthalpy from the pool surface to the containment atmosphere. As the containment atmosphere heats up, it pressurizes, but the containment pressurization rates are very different for LOCA and ATWS. ATWS containment pressurization will occur at a slower rate.

Typically, for LOCA analysis, conservative assumptions are made to calculate the containment pressure. These assumptions tend to drive the containment pressure higher. However, when containment over-pressure credit is required to ensure that ECCS equipment satisfies operability limits (e.g., NPSH), the above assumptions are not conservative, because the calculation predicts a pressure higher than expected.

Thus, when containment over-pressure credit is taken for ECCS equipment, the containment calculations must use modeling assumptions that slow the pressurization rate. The LOCA containment conservative assumptions are likely not conservative for ATWS containment over-pressure credits.

9.3.1.2.6 Effect of MELLLA+ on Suppression Pool Temperatures and NPSH Requirements of Critical Equipment

The calculated suppression pool temperatures indicate that NPSH limits are likely to be violated during an ATWS event. All the modeled full isolation ATWS events initiating at the MELLLA+ corner (120 percent OLTP, 80 percent flow) reach the HCTL before boron injection can shutdown the reactor. Therefore, emergency depressurization is more likely under MELLLA+ conditions than at OLTP for a full isolation ATWS on plants with stand-pipe boron injection. On plants with boron injection through the CS, the boron is very effective shutting down the reactor early, and HCTL is not likely to be reached.

Following the depressurization, the reactor may remain in a shutdown condition, it can regain criticality and remain at a low power, or it may re-pressurize. The outcome is uncertain on a generic basis and needs to be evaluated on a plant-specific basis. Simulations show that in all cases where the reactor needs to be depressurized, the suppression pool temperature reaches high temperatures. In some of the cases where re-pressurization occurs, the calculated suppression pool enthalpy may cause containment pressures to rise above the limits.

The effect on NPSH requirements will be plant-dependent. For example, plants that inject boron through the CS will shutdown promptly and the final suppression pool temperature is likely to be small enough the all ECCS equipment will be well within NPSH limits. Large plants with small containment that inject boron through the stand pipes will likely have to depressurize if operating at the MELLLA+ corner and some equipment may reach NPSH limits.

The NRC staff concludes that this issue cannot be resolved generically. For each individual MELLLA+ application, the licensee must demonstrate the expected ATWS performance of its plant and evaluate the impact if any of NPSH limits on ECCS equipment performance. The plant-specific ATWS calculations must take into account the operability limits (e.g., NPSH) for all ECCS equipment assumed available for the calculation. See ATWS LOOP limitation in Section 9.3.1.1.

9.3.1.2.7 Effect of MELLLA+ on Availability of Sufficient Volume of High Pressure Injection

MELLLA+ increases the effective operating rod line. During an ATWS event, following the prescribed recirculation pump trip, the reactor will settle at a higher power than for pre-MELLLA+/EPU conditions. This higher power will result in increased steam flow and higher requirement of high pressure injection water volumes in order to maintain the desired water level control. MELLLA+ applicants should verify that the available high pressure injection sources provide sufficient volume at a rate that will maintain the target water level specified in the plant-specific emergency operating procedures (EOPs).

The plant-specific ATWS calculations should be reviewed to ensure that the maximum available injection volume is sufficient to maintain the water level strategy. Note that different ATWS scenarios result in different injection systems being available. For example, if MSIV's are open, some plants would allow the use of the feedwater systems, which have sufficient discharge volume. But if MSIV's are closed, some plants will loose all feedwater pumping capacity and

must rely on other sources like High Pressure Coolant Injection (HPCI), which may or may not have sufficient volume and injection rate to maintain the EOP target water level.

The results of this evaluation are highly plant-specific because they depend on specific balance of plant hardware configurations and capability, which vary widely through the fleet. The staff review must ensure that the analysis assumptions are consistent with the plant configuration.

9.3.1.3 Plant-Specific ATWS Analyses

ODYN is the approved licensing code for ATWS calculations, but ODYN is only licensed for water level strategies at TAF+5 or above due to some modeling limitations. In addition, ODYN cannot model a reactor depressurization. Therefore, ODYN cannot model all the mitigation actions required by the EOPs, especially lower water levels or depressurization. In addition, the NRC staff's confirmatory calculations show that ODYN is not conservative in terms of suppression pool temperature throughout the scenario timeline, but only at the end of the transient.

LTR NEDC-33006P (Reference 1) states that for plant-specific calculations, "The ATWS evaluation will be performed using the approved ODYN methodology documented in Section 5.3.4 of ELTR1. The ATWS analysis using the ODYN methodology will remain as the plant's licensing basis; however, a best estimate TRACG analysis will be performed to support NRC review for those plants that have EOPs requiring depressurization prior to the plant achieving hot shutdown. The TRACG analysis is performed consistent with the assumed operator actions, including depressurization, for ATWS with isolation scenarios. The operator actions would be consistent with the NRC-approved EPGs and EOPs and the basis for operator action assumptions will be described in the M+SAR. The transient duration of the TRACG modeling the depressurization scenario will continue until the power is effectively suppressed by boron injection." The NRC staff concludes that this approach is an acceptable implementation given the following limitation.

ATWS TRACG Analysis Limitation

a) For plants that do not achieve hot shutdown prior to reaching the heat capacity temperature limit (HCTL) based on the licensing ODYN code calculation, plant-specific MELLLA+ implementations must perform best-estimate TRACG calculations on a plant-specific basis. The TRACG analysis will account for all plant parameters, including water-level control strategy and all plant-specific emergency operating procedure (EOP) actions.

b) The TRACG calculation is not required if the plant increases the boron-10 concentration/enrichment so that the integrated heat load to containment calculated by the licensing ODYN calculation does not change with respect to a reference OLTP/75 percent flow ODYN calculation.

c) Peak cladding temperature (PCT) for both phases of the transient (initial overpressure and emergency depressurization) must be evaluated on a plant-specific basis with the TRACG ATWS calculation.

d) In general, the plant-specific application will ensure that operation in the MELLLA+ domain is consistent with the assumptions used in the ATWS analysis, including equipment out of service (e.g., FWHOOS, SLO, SRVs, SLC pumps, and RHR pumps, etc.). If assumptions are not satisfied, operation in MELLLA+ is not allowed. The SRLR will specify the prohibited flexibility options for plant-specific MELLLA+ operation, where applicable. For key input parameters, systems and engineering safety features that are

important to simulating the ATWS analysis and are specified in the Technical Specification (TS) (e.g., SLCS parameters, ATWS RPT, etc.), the calculation assumptions must be consistent with the allowed TS values and the allowed plant configuration. If the analyses deviate from the allowed TS configuration for long term equipment out of service (i.e., beyond the TS LCO), the plant-specific application will specify and justify the deviation. In addition, the licensee must ensure that all operability requirements are met (e.g., NPSH) by equipment assumed operable in the calculations.

e) Nominal input parameters can be used in the ATWS analyses provided the uncertainty treatment and selection of the values of these input parameters are consistent with the input methods used in the original GE ATWS analyses in NEDE-24222. Treatment of key input parameters in terms of uncertainties applied or plant-specific TS value used can differ from the original NEDE-24222 approach, provided the manner in which it is used yields more conservative ATWS results.

f) The plant-specific application will include tabulation and discussion of the key input parameters and the associated uncertainty treatment.

TRACG is not currently licensed for ATWS calculations. In MFN 07-034 (Reference 27), GHNE stated that TRACG will be submitted for NRC staff review and approval for all ATWS scenarios. TRACG is a best estimate code that can model the ATWS scenario with more fidelity, including all the required operator actions and water level strategies. It is noted that the need for depressurization may not be limited to plants operating at EPU/MELLLA+. Plants operating at or above the MELLLA rod line may experience the need to depressurize reactor if the suppression pool reaches the HCTL, even if these plants meet the specific set of requirements stipulated in 10 CFR 50.62. Thus, the NRC staff recommends the use of ODYN licensing calculations, to be supplemented by TRACG best-estimate confirmatory calculations that include all operator actions for MELLLA+ implementation.

9.3.1.4 ATWS Conclusion

The NRC staff has reviewed the information submitted related to ATWS and concludes that GHNE has adequately accounted generically for most effects of the proposed MELLLA+/EPU operation on ATWS. However, the NRC staff concludes that the MELLLA+ operation affects the reactor's ATWS performance and the results of generic calculations show significant variability. Therefore, the NRC staff requires best-estimate ATWS TRACG calculations on a plant-specific basis, which account for all plant parameters, including water-level control strategy, all plant-specific EOP actions, and EOOS allowed by TSs, to demonstrate compliance with the requirements of 10 CFR 50.62.

The NRC staff strongly recommends that licensees that plan to implement MELLLA+/EPU should increase the boron concentration of the SLCS so that the integrated heat load to containment remains constant. For example, if the power density is increased by 10 percent, the boron injection time must be reduced by 10 percent so the integrated heat load remains constant.

9.3.2 SBO

The NRC staff reviewed Section 9.3.2 of LTR NEDC-33006P to verify the GE's contention that the plant response to and coping capabilities for the SBO event are not affected by operation in the MELLLA+ CF range. Based on this review, the NRC staff agrees that the plant response to and coping capabilities for the SBO event will not be affected by the proposed expansion of the

power/flow map, because there is no change in the core power, decay heat, pressure, or steam flow as a result of the MELLLA+ operating range expansion. Therefore, the NRC staff concludes that the implementation of MELLLA+ is acceptable for the SBO event.

9.3.3 ATWS with Core Instability

ATWS is defined as an AOO followed by the failure of the reactor portion of the protection system specified in GDC 20. ATWS-Stability is defined as an ATWS event where large amplitude unstable power oscillations develop.

The NRC staff evaluated the potential for thermal-hydraulic instability in conjunction with ATWS events using the methods and criteria approved by the NRC staff. For this analysis, the NRC staff reviewed the limiting event determination, the sequence of events, the analytical model and its applicability, the values of parameters used in the analytical model, and the results of the analyses. Review guidance is provided in SRP Section 15.8 and Matrix 8 Table of RS-001.

9.3.3.1 ATWS with Core Instability Assessment

There are an unlimited number of ATWS scenarios. Of all those scenarios, one-class ATWS events result in unstable power oscillations of extremely large amplitude. These events have in common an unlimited supply of very cold, unheated, water being pumped into the vessel because cold condenser water is available, but turbine extraction steam is not. The cold water supply increases the subcooling, which increases very significantly the reactor power. The resulting low-flow, high-power conditions cause the instability. This class of events is generically referred as "ATWS-Stability."

ATWS-Stability was found to be unacceptable even at OLTP. Extremely large power oscillations (greater than 1000 percent) develop during this events and fuel integrity is compromised. The "ATWS-Stability Mitigation Actions" were developed to mitigate the consequences of these instabilities during ATWS. The mitigation actions were included in Revision 4 of the EPGs in the early 1990's and, now, form part of the EOPs for every operating BWR. The most relevant EPG mitigation actions are: (1) early water level reduction to 2 feet below the feed-water sparger, and (2) early boron injection. The EPG mitigation actions were found to be effective when operating at OLTP in suppressing the unstable oscillations and their negative consequences.

The NRC staff review indicates that, in principle, MELLLA+ operation affects ATWS-Stability. Operation in the MELLLA+ corner (120 percent OLTP, 80 percent CF) results in the reactor settling at significantly higher power, following a RPT, than the event initiated at rated OLTP condition. This higher power makes the final power even larger after the feedwater cool-down period. Thus, at least in principle, the unstable power oscillations are enhanced under MELLLA+, and their consequences should be more severe. This effect is illustrated in Figure 9-1.

Table 9-5 of NEDC-33006P documents a series of simulations of ATWS/Stability events without the prescribed EPG mitigation actions. For these hypothetical situations, the oscillations grow quite large and fuel integrity is compromised (PCT >2200F) in three of the five simulations.

TRACG simulations performed by GHNE have demonstrated that the EPG mitigation actions are still effective in suppressing the oscillations during these classes of ATWS events. Figure 9-19 shows the evolution of an ATWS-Stability event without mitigation actions. The unstable power oscillations are allowed to grow to greater than 1000 percent. Following one of the power

excursions, the fuel dries out and fails to re-wet. The resulting temperature excursion is sufficiently large to compromise the integrity of the fuel.

Figure 9-20 shows the evolution of the same ATWS-Stability event as in Figure 9-18, but following the prescribed EPG Mitigation Actions, which include injection of boron and water level reduction once the unstable power oscillations are identified. As observed, the power oscillations are managed adequately by the early mitigation actions and fuel integrity is not challenged. The most effective mitigation action is the water level reduction. Early boron injection helps the transient, but it is too slow to mitigate the oscillations. Figure 9-21 shows an ATWS/Stability event with only boron injection. As seen in this figure, the large amplitude oscillations remain for a significantly longer period of time than when the water level is lowered.

However, the EPG mitigation actions are manual operator actions. For all these simulations, any operator are delayed 120 seconds to account for variability of human response. As seen in Figure 9-20, the unstable power oscillations can grow quite large before the mitigation actions become effective.

The amplitude of the oscillations before the mitigation actions become effective is likely to have a large sensitivity to specific plant parameters. Specifically, there is a large sensitivity to the characteristics of the feedwater system, which drive the event. Therefore, these calculations will be repeated on a plant-specific bases to demonstrate that the unstable power oscillations do not grow sufficiently large to challenge fuel integrity before the mitigation actions suppress them.

Plant-Specific ATWS Instability Limitation

Until such time that NRC approves a generic solution for ATWS instability calculations for MELLLA+ operation, each plant-specific MELLLA+ application must provide ATWS instability analysis that satisfies the ATWS acceptance criteria listed in SRP Section 15.8. The plant-specific ATWS instability calculation must: (1) be based on the peak-reactivity exposure conditions, (2) model the plant-specific configuration important to ATWS instability response including mixed core, if applicable, and (3) use the regional-mode nodalization scheme. In order to improve the fidelity of the analyses, the plant-specific calculations should be based on latest NRC-approved neutronic and thermal-hydraulic codes such as TGBLA06/PANAC11 and TRACG04.

Generic ATWS Instability Limitation

Once the generic solution is approved, the plant-specific applications must provide confirmation that the generic instability analyses are relevant and applicable to their plant. Applicability confirmation includes review of any differences in plant design or operation that will result in significantly lower stability margins during ATWS such as:

- turbine bypass capacity,
- fraction of steam-driven feedwater pumps,
- any changes in plant design or operation that will significantly increase core inlet subcooling during ATWS events,
- significant differences in radial and axial power distributions,
- hot-channel power-to-flow ratio,
- fuel design changes beyond GE14

9.3.3.2 ATWS with Core Instability Conclusion

The NRC staff concludes that operation in the MELLLA+ domain does not significantly reduce ATWS-stability safety margins when the operator follows the EPGs, including the stability mitigation actions, in a timely fashion.

Table 9-11 Summary of TRACG AOO ΔCPR/ICPR Results from NEDC-32906P

Event	105 % OLTP 110 % Core Flow	105 % OLTP 100 % Core Flow	105 % OLTP 75 % Core Flow
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			]]

Table 9-12 Non-Mitigated ATWS Instability Limiting Fuel Conditions for Bounding Turbine Trip with Full Bypass Event [1]

Fuel Type	Initial Operating Statepoint (%OLTP, % Rated Core Flow)	Initial Core Power to Flow Ratio (MW/Mlb/hr)	Oscillation Mode	PCT (°K/°F)	Maximum Power Spike Energy Deposition (cal/g)
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Table 9-13 Key Event Timing for an MSIV Closure ATWS Case

Event	Time (sec)
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Table 9-14 Containment parameters used for suppression pool calculations

Parameter	BRUNSWICK / MSIVC / 120 %P, 85 % F	CLINTON / MSIVC / 120%P, 85 % F	Browns Ferry / MSIVC / 120 %P, 85 % F
Initial Suppression Pool Temperature (EF)	95	95	95
Initial Suppression Pool Mass (lbm)	5,364,900	8,204,000	7,626,000
RHR Service Water Temperature (EF)	92	95	95
RHR Heat Exchanger K-Factor per Loop in Containment Cooling Mode (Btu/sec-EF)	235	360	223

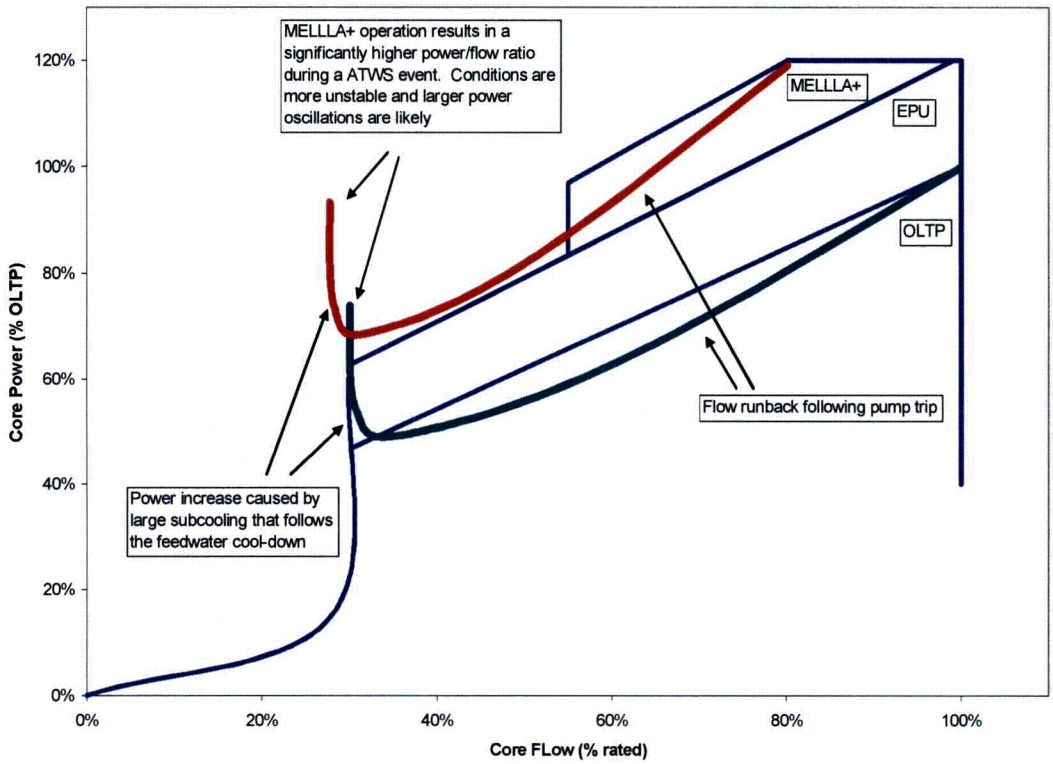

Figure 9-1 Illustration of reactor power during ATWS events from MELLLA+ and OLTP initial conditions. The final power is significantly larger under MELLLA+

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Figure 9-2 Comparison of suppression pool temperature calculated by ODYN and by TRACG following the EOPs, including depressurization

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Figure 9-3 Comparison of pressure response for MELLLA+ and OLTP for MSIV Closure with one SRV out of service

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Figure 9-4 Suppression pool temperature calculated for several reactors

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Figure 9-5 TRACG power response for ATWS with water level reduction to TAF showing recriticality at 1200 to 1500 seconds

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Figure 9-6 Peak fuel clad temperature calculated by TRACG for isolation ATWS. During emergency de-pressurization, the core uncovers and clad over-heating occurs, but clad temperature criteria are satisfied.

[[

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Figure 9-7 TRACG pressure response for ATWS with water level reduction to TAF+5' showing recriticality at 1200 to 1500 seconds

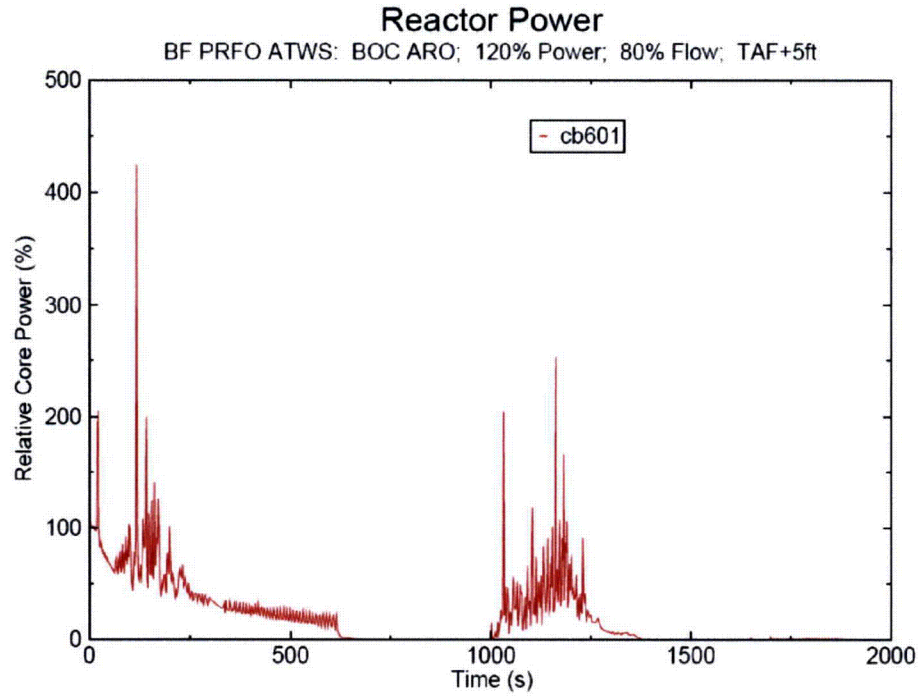

Figure 9-8 Reactor power calculated by TRACE for isolation ATWS.

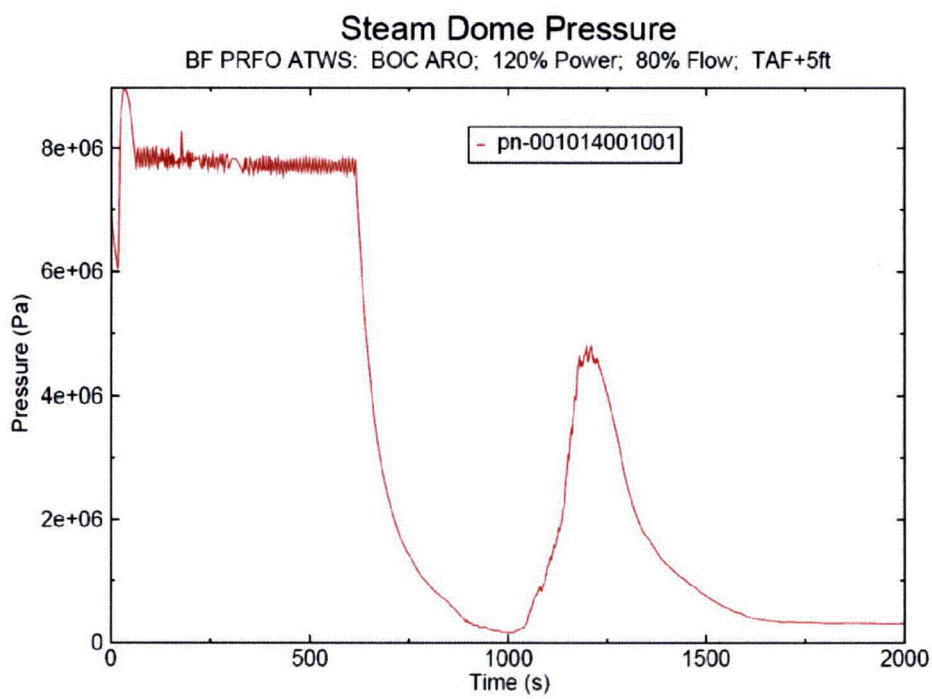

Figure 9-9 Reactor pressure calculated by TRACE for an isolation ATWS.

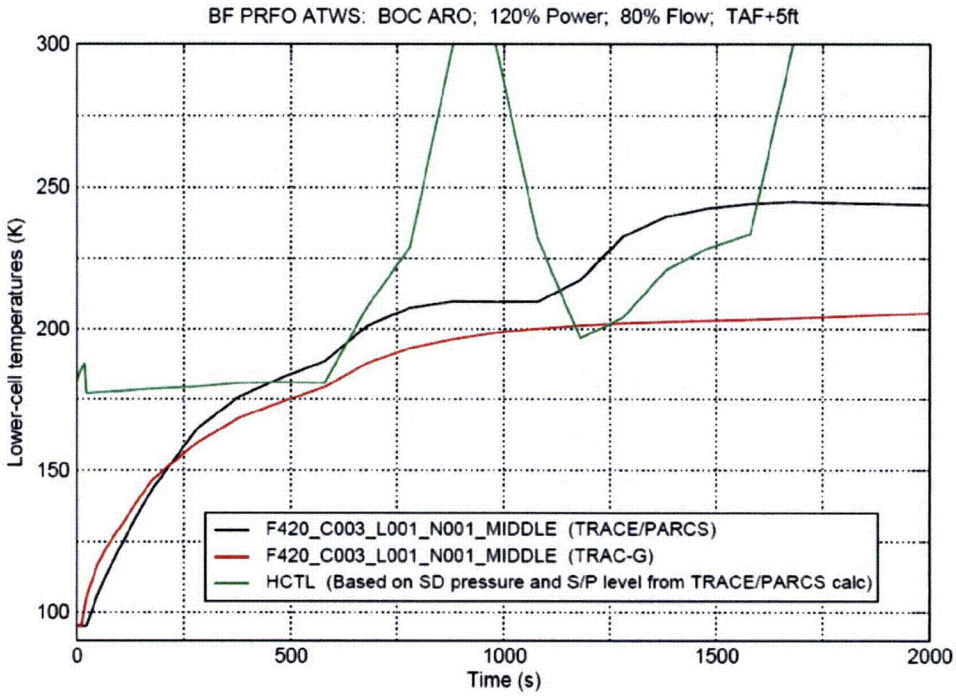

Figure 9-10 Suppression pool temperatures calculated for the above isolation ATWS by TRACG and TRACE.

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Figure 9-11 ATWS - MSIVC followed by RPT

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Figure 9-12 Reactor Power Response for 3 Water Level Control Strategies

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Figure 9-13 SRV Flow Rate Response for 3 Water Level Control Strategies

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Figure 9-14 Axial PCT Response at Different Times

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Figure 9-15 Boron Flux in Upper Region of Lower Plenum – Ring 1

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Figure 9-16 Boron Flux in Upper Region of Lower Plenum – Ring 2

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Figure 9-17 Boron Concentration in Upper Region of Lower Plenum – Ring 1

[[

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Figure 9-18 Boron Concentration in Upper Region of Lower Plenum – Ring 2

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Figure 9-19 Unstable power oscillations reached ~1000 percent during an ATWS-Stability event that does no follow the prescribed EPG Mitigation Actions. Fuel integrity is compromised by failure to re-wet and the associated temperature excursion

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Figure 9-20 When the prescribed EPG Mitigation Actions are followed, the ATWS/Stability event is managed without compromising fuel integrity

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Figure 9-21. Partially Mitigated ATWS/Stability event. Only Boron injection without water level reduction

10.0 OTHER EVALUATIONS

This section provides the NRC staff review of the reactor safety performance evaluations. Table 10-1 lists the specific topics and the corresponding NRC staff dispositions for plant-specific application.

Table 10-1 Other Evaluation Topics

Section	Title	[[	
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		
10.9	Emergency and Abnormal Operating Procedures		]]

Plant-specific evaluations will be included in the plant-specific submittal consistent with the format and level of detail as discussed in LTR NEDC-33006P 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.

10.1 HIGH ENERGY LINE BREAK (HELB)

HELBs are evaluated for their effects on equipment qualification. [[

]] The scope of these evaluations includes MELLLA+ 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 (MELB)

[[

]] Therefore, the MELBs will not effect environmental qualification and is not evaluated on a plant-specific basis.

10.3 ENVIRONMENTAL QUALIFICATION

Safety related components are required to be qualified for the environment in which they are required to operate. There is no change or increase in core power, radiation levels, decay heat, pressure, steam flow, feedwater flow, normal process temperatures, pressures, or flow rates as a result of the MELLLA+ operating range expansion. The change in fluid induced loads on safety-related components is discussed in Sections 3.2.2, 3.5, and 4.1.2 of this SE.

10.3.1 Electrical Equipment

There is no change in core power, radiation levels, decay heat, pressure, steam flow, or feedwater flow as a result of the MELLLA+ operating range expansion. [[

]] Therefore, there is no change to the environmental qualification (EQ) for safety related electrical equipment located inside or outside of containment.

10.3.2 Mechanical Equipment With Non-Metallic Components

There is no change to the EQ for safety related mechanical equipment with non-metallic components located inside or outside of containment, because operation in the MELLLA+ operating range does not increase any of the normal process temperatures or the normal and accident radiation levels. Therefore, NRC staff finds the LTR evaluation of EQ for safety related mechanical equipment with non-metallic components to be acceptable.

10.3.3 Mechanical Component Design Qualification

The mechanical design of equipment/components (e.g., heat exchangers) is not affected by operation in the MELLLA+ operating range, [[

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The change in fluid induced loads on safety-related components is discussed in Sections 3.2.2, 3.5, and 4.1.2. The mechanical components and component supports are adequately designed for the MELLLA+ operating range, [[

]] Therefore, NRC staff finds the LTR evaluation of mechanical component design qualification to be acceptable.

10.4 TESTING

When the MELLLA+ operating range expansion is implemented, plant-specific testing will be performed to confirm operational performance and control aspects of the MELLLA+ changes. The NRC staff finds this to be acceptable since it ensures that the testing will be performed and that it will be appropriate for every plant proposing to use MELLLA+.

10.5 INDIVIDUAL PLANT EVALUATION

LTR NEDC-33006P is not risk-informed, but does include in Section 10.5 a discussion of risk-related factors to be considered in the plant-specific M+SAR. GE presents a generic discussion of these factors, which includes initiating event categories and frequency, component reliability, operator response, success criteria, external events, shutdown risk, and probabilistic risk assessment (PRA) quality. GE concludes that there are no significant effects of MELLLA+ on these risk-related topics and that analysis of plants that have uprated to power levels up to 120 percent of OLTP indicate that the incremental risk increase due to MELLLA+ operating range expansion will be negligible relative to the risk increase associated with their EPU. Finally, GE states that the key inputs to the plant-specific risk that support the [[ ]] will be confirmed in the licensee's plant-specific M+ SAR.

Since LTR NEDC-33006P is not risk-informed and the key inputs to the plant-specific risk that support the [[ ]] will be confirmed in the plant-specific M+SAR, the NRC has not performed an in-depth review of the LTR's risk discussion and has not relied upon this information in determining the acceptability of the LTR. Thus, the NRC has not made a [[ ]] Therefore, the following limitation applies:

Individual Plant Evaluation Limitation

Licensees that submit a MELLLA+ application should address the plant-specific risk impacts associated with MELLLA+ implementation, consistent with approved guidance documents (e.g., NEDC-32424P-A, NEDC-32523P-A, and NEDC-33004P-A) and the Matrix 13 of RS-001 and re-address the plant-specific risk impacts consistent with the approved guidance documents that were used in their approved EPU application and Matrix 13 of RS-001. If an EPU and MELLLA+ application come to the NRC in parallel, the expectation is that the EPU submittal will have incorporated the MELLLA+ impacts.

10.5.1 Operator Response

The LTR stated that the operator responses to anticipated transients, emergency, and special events (such as ATWS) with the plant operation of MELLLA+ for GE BWRs with EPU conditions are the same as plants that currently operate under EPU conditions. The justification is that the long term cooling evaluated for EPU conditions is not impacted by MELLLA+ operating range expansion. The operator actions generically described for EPU conditions remain similar to those actions described in GHNE's LTR, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," for emergency conditions.

The NRC staff inquired in the May 31st teleconference if GHNE could identify any scenarios, such as ATWS, where the available times of operator response could be adversely affected due to MELLLA+. GHNE responded that operator responses are expected to be similar to what licensees use for EPU conditions. However, GHNE stated in the LTR and in the teleconference that licensees are expected to provide individual bases for specified operator response times in which they identified as being adversely impacted by MELLLA+ operation. The NRC staff did not identify any critical operator actions being adversely affected by plant operation in the MELLLA+ regions and agrees with GHNE assessment that individual licensees are expected to identify and provide justification for operator response times, including those operator actions that could exceed the existing available times for certain emergency scenarios. The NRC staff finds GHNE evaluation of operator actions under MELLLA+ acceptable.

10.6 OPERATOR TRAINING AND HUMAN FACTORS

The NRC's acceptance criteria for human factors are based on Title 10 to the *Code of Federal Regulations* (10 CFR), Part 50, Appendix A, General Design Criterion (GDC)-19, "Control room," 10 CFR 50.120, "Training and qualification fo nuclear power plant personnel," 10 CFR Part 55, "Operators' Licenses," and the guidance in Generic Letter (GL) 82-33, "Supplement 1 to NUREG-0737, Requirements for Emergency Response Capability," dated December 17, 1982. Specific review criteria are contained in the NUREG-0800 (Revision 1) "Standard Review Plan," Sections 13.2.1, 13.2.2, 13.5.2.1, and Chapter 18.0.

GHNE stated in the LTR that licensees will be expected to state the following items regarding control room changes in their SEs for plant operation in MELLLA+:

- Identification of control room modifications needed to support MELLLA+
- Definition of a schedule for changes in control room displays, controls, and alarms to be implemented in preparation for MELLLA+ operation
- Explanation of how operators will be trained on the control room modifications

The NRC staff agrees that individual licensees will have to identify all modifications needed to support MELLLA+ as well as the licensees providing a schedule and operator training of these modifications before operating in MELLLA+. The NRC staff finds GHNE's evaluation of potential control room changes to be acceptable.

GHNE stated that operator training will be conducted by individual licensees prior to operation of the unit in the MELLLA+ region. Licensees can obtain data during operation in the MELLLA+ region for incorporation into operator training as needed. The classroom training will cover various aspects of MELLLA+, including changes to the power/flow map, changes to important setpoints, plant procedures, and startup test procedures. The classroom training may be combined with simulator training for operational sequences that are unique to MELLLA+. GHNE stated that simulator training on existing transients should not be anticipated since the plant dynamics will not change substantially for operation in the MELLLA+ region. However, licensees should perform simulator changes and fidelity validation in accordance with applicable American National Standards Institute standards currently being used for training simulators.

The NRC staff agrees that operator training and simulator modifications and validations should be handled on a plant-specific basis. Licensees will be required to implement operator training on any changes made to operator actions, emergency operating procedures (EOPs) and abnormal operating procedures (AOPs), and control room components due to MELLLA+ to ensure that operators are aware of those changes. Also, any modifications that are needed for control room simulators are expected to be made using standards and methods previously accepted by the NRC staff. The NRC staff finds GHNE evaluations on operator training and simulator related to plant operation in MELLLA+ acceptable.

10.7 PLANT LIFE

With regard to increasing the potential of the core internals to be affected by irradiation assisted stress corrosion cracking (IASCC), GE states that a slight increase in peak fluence experienced by the reactor internals may cause a minor increase in the potential for IASCC. However, the current inspection strategy for the reactor internals components is adequate to manage any

potential effects of MELLLA+, and that a plant-specific assessment for MELLLA+ will be provided for each plant.

For reactor internals and core support materials, note 1 in the Matrix 1 Table in the Office of Nuclear Reactor Regulation RS-001, "Review Standard for Extended Power Uprates," states that guidance on the neutron irradiation-related threshold for inspection for irradiation-assisted stress-corrosion cracking for BWRs is in the Boiling Water Reactors Vessel Integrity Program (BWRVIP)-26. BWRVIP-26 indicates that components receiving a neutron radiation fluence greater than 5x1020 n/cm2 (E>1MeV) are susceptible to IASCC. Licensees that utilize LTR NEDC-33006P must provide a plant-specific IASCC evaluation when implementing MELLLA+, which includes:

1. the components that will exceed the IASCC threshold of 5x1020 n/cm2 (E>1 MeV),

2. the impact of failure of these components on the integrity of the reactor internals and core support structures under licensing design bases conditions, and

3. the inspections that will be performed on components that exceed the IASCC threshold to ensure timely identification of IASCC, should it occur.

The NRC staff concludes that a plant-specific assessment for MELLLA+ which includes the proper inspection programs with the above MELLLA+ plant-specific action items will provide the proper assurance that degradation is promptly identified and corrected so that the safety-related reactor internals will continue to perform in service as designed.

The NRC staff's review of LTR NEDC-33006P, Section 10.7, indicates that the methods and analyses in the LTR are generally acceptable. The NRC staff finds that, with the addition of the limitation below, GE has provided adequate specific direction to the BWR licensees for assessing the impact of MELLLA+ on its facility. Therefore, the NRC staff concludes that a licensee's adherence to the requirements of the LTR, and the completion of the limitation below, will facilitate future NRC staff reviews of MELLLA+ licensing amendment requests. This LTR may be used as a reference for implementing MELLLA+, concerning these sections in a license amendment for GE designed BWRs to the extent specified and under the limitations delineated in the LTR and in this SE.

IASCC Limitation

The applicant is to provide a plant-specific IASCC evaluation when implementing MELLLA+, which includes the components that will exceed the IASCC threshold of $5x10^{20}$ n/cm^2 (E>1MeV), the impact of failure of these components on the integrity of the reactor internals and core support structures under licensing design bases conditions, and the inspections that will be performed on components that exceed the IASCC threshold to ensure timely identification of IASCC, should it occur.

10.9 EMERGENCY AND ABNORMAL OPERATING PROCEDURES

GHNE stated that EOPs and AOPs can be affected by MELLLA+ operation. The changes to the EOPs will include revised variables and limit curves, which define conditions where operator actions are indicated. The Safety Parameter Display System will also be updated along with the EOPs in these areas. The AOPs will be reviewed for MELLLA+ impact on event-based operator actions. GHNE stated that the operator actions would likely remain the same and no new actions would be expected; however, individual licensees are advised to review all AOPs for confirmation and revise as necessary before implementation of MELLLA+.

The NRC staff takes into consideration that all plants have EOPs and AOPs that are plant-specific and licensees would be expected to indicate any EOP and AOP changes related to MELLLA+ operation as described by GHNE in their amendment requests. Therefore, the NRC staff finds GHNE assessment of potential EOP and AOP changes related to MELLLA+ acceptable.

11.0 LICENSING EVALUATIONS

The NRC staff will evaluate the technical specification changes, the environmental assessment, and the significant hazards consideration assessment on a plant-specific basis.

12.0 LIMITATIONS AND CONDITIONS

12.1 GEXL-PLUS (SECTION 1.1.4)

The plant-specific application will confirm that for operation within the boundary defined by the MELLLA+ upper boundary and maximum CF range, the GEXL-PLUS experimental database covers the thermal-hydraulic conditions the fuel bundles will experience, including, bundle power, mass flux, void fraction, pressure, and subcooling. If the GEXL-PLUS experimental database does not cover the within bundle thermal-hydraulic conditions, during steady state, transient conditions, and DBA conditions, GHNE will inform the NRC at the time of submittal and obtain the necessary data for the submittal of the plant-specific MELLLA+ application.

In addition, the plant-specific application will confirm that the experimental pressure drop database for the pressure drop correlation covers the pressure drops anticipated in the MELLLA+ range.

With subsequent fuel designs, the plant-specific applications will confirm that the database supporting the CPR correlations covers the powers, flows and void fractions BWR bundles will experience for operation at and within the MELLLA+ domain, during steady state, transient, and DBA conditions. The plant-specific submittal will also confirm that the NRC staff reviewed and approved the associated CPR correlation if the changes in the correlation are outside the GESTAR II (Amendment 22) process. Similarly, the plant-specific application will confirm that the experimental pressure drop database does cover the range of pressures the fuel bundles will experience for operation within the MELLLA+ domain.

12.2 RELATED LTRS (SECTION 1.1.5)

Plant-specific MELLLA+ applications must comply with the limitations and conditions specified in and be consistent with the purpose and content covered in the NRC staff SEs approving the latest version of the following LTRs: NEDC-33173P, NEDC-33075P, and NEDC-33147 (References 37, 45, and 47).

12.3 CONCURRENT CHANGES (SECTION 1.2.1)

a) The plant-specific analyses supporting MELLLA+ operation will include all operating condition changes that are implemented at the plant at the time of MELLLA+ implementation. Operating condition changes include, but are not limited to, those

changes that affect, an increase in the dome pressure, maximum CF, fuel cycle length, or any changes in the licensed operational enhancements. For example, with an increase in dome pressure, the following analyses must be analyzed: the ATWS analysis, the ASME overpressure analyses, the transient analyses, and the ECCS-LOCA analysis. Any changes to the safety system settings or any actuation setpoint changes necessary to operate with the increased dome pressure must be included in the evaluations (e.g., SRV setpoints).

b) For all topics in LTR NEDC-33006P that are reduced in scope or generically dispositioned, the plant-specific application will provide justification that the reduced scope or generic disposition is applicable to the plant. If changes that invalidate the LTR dispositions are to be implemented at the time of MELLLA+ implementation, the plant-specific application will provide analyses and evaluations that demonstrate the cumulative effect with MELLLA+ operation. For example, if the dome pressure is increased, the ECCS performance will be evaluated on a plant-specific basis.

c) Any generic bounding sensitivity analyses provided in LTR NEDC-33006P will be evaluated to ensure that the key plant-specific input parameters and assumptions are applicable and bounded. If these generic sensitivity analyses are not applicable or additional operating condition changes affect the generic sensitivity analyses, a plant-specific evaluation will be provided. For example, with an increase in the dome pressure, the ATWS sensitivity analyses that model operator actions (e.g., depressurization if the HCTL is reached) needs to be reanalyzed, using the bounding dome pressure condition.

d) If a new GE fuel product line or another vendor's fuel is loaded at the plant, the applicability of any generic sensitivity analyses supporting the MELLLA+ application shall be justified in the plant-specific application. If the generic sensitivity analyses cannot be demonstrated to be applicable, the analyses will be performed including the new fuel. For example, the ATWS instability analyses supporting the MELLLA+ condition are based on the GE14 fuel response. New analyses that demonstrate the ATWS instability performance of the new GE fuel or another vendor's fuel for MELLLA+ operation shall be provided to support the plant-specific application.

e) If a new GE fuel product line or another vendor's fuel is loaded at the plant prior to a MELLLA+ application, the analyses supporting the plant-specific MELLLA+ application will be based on a specific core configuration or bounding core conditions. Any topics that are generically dispositioned or reduced in scope in LTR NEDC-33006P will be demonstrated to be applicable, or new analyses based on the specific core configuration or bounding core conditions will be provided.

f) If a new GE fuel product line or another vendor's fuel is loaded at the plant prior to a MELLLA+ application, the plant-specific application will reference an NRC-approved stability method supporting MELLLA+ operation, or provide sufficient plant-specific information to allow the NRC staff to review and approve the stability method supporting MELLLA+ operation. The plant-specific application will demonstrate that the analyses and evaluations supporting the stability method are applicable to the fuel loaded in the core.

g) For MELLLA+ operation, core instability is possible in the event a transient or plant maneuver places the reactor at a high power/low-flow condition. Therefore, plants operating at MELLLA+ conditions must have a NRC-approved instability protection method. In the event the instability protection method is inoperable, the applicant must employ an NRC-approved backup instability method. The licensee will provide technical

specification (TS) changes that specify the instability method operability requirements for MELLLA+ operation, including any backup stability protection methods.

12.4 RELOAD ANALYSIS SUBMITTAL (SECTION 1.2.2.3.2)

The plant-specific MELLLA+ application shall provide the plant-specific thermal limits assessment and transient analysis results. Considering the timing requirements to support the reload, the fuel- and cycle-dependent analyses including the plant-specific thermal limits assessment may be submitted by supplementing the initial M+SAR. Additionally, the SRLR for the initial MELLLA+ implementation cycle shall be submitted for NRC staff confirmation.

12.5 OPERATING FLEXIBILITY (SECTION 1.3.3)

a) The licensee will amend the TS LCO for any equipment out-of-service (i.e., SLO) or operating flexibilities prohibited in the plant-specific MELLLA+ application.

b) For an operating flexibility, such as FWHOOS, that is prohibited in the MELLLA+ plant-specific application but is not included in the TS LCO, the licensee will propose and implement a license condition.

c) The power flow map is not specified in the TS; however, it is an important licensed operating domain. Licensees may elect to be licensed and operate the plant under plant-specific-expanded domain that is bounded by the MELLLA+ upper boundary. Plant-specific applications approved for operation within the MELLLA+ domain will include the plant-specific power/flow map specifying the licensed domain in the COLR.

12.6 SLMCPR STATEPOINTS AND CF UNCERTAINTY (SECTION 2.2.1.1)

Until such time when the SLMCPR methodology (References 40 and 41) for off-rated SLMCPR calculation is approved by the staff for MELLLA+ operation, the SLMCPR will be calculated at the rated statepoint (120 percent P/100 percent CF), the plant-specific minimum CF statepoint (e.g., 120 percent P/80 percent CF), and at the 100 percent OLTP at 55 percent CF statepoint. The currently approved off-rated CF uncertainty will be used for the minimum CF and 55 percent CF statepoints. The uncertainty must be consistent with the CF uncertainty currently applied to the SLO operation or as NRC-approved for MELLLA+ operation. The calculated values will be documented in the SRLR.

12.7 STABILITY (SECTION 2.4.1)

Manual operator actions are not adequate to control the consequences of instabilities when operating in the MELLLA+ domain. If the primary stability protection system is declared inoperable, a non-manual NRC-approved backup protection system must be provided, or the reactor core must be operated below a NRC-approved backup stability boundary specifically approved for MELLLA+ operation for the stability option employed.

12.8 FLUENCE METHODOLOGY AND FRACTURE TOUGHNESS (SECTION 3.2.1)

The applicant is to provide a plant-specific evaluation of the MELLLA+ RPV fluence using the most up-to-date NRC-approved fluence methodology. This fluence will then be used to provide a plant-specific evaluation of the RPV fracture toughness in accordance with RG 1.99, Revision 2.

12.9 REACTOR COOLANT PRESSURE BOUNDARY (SECTION 3.5.1)

MELLLA+ applicants must identify all other than Category "A" materials, as defined in NUREG-0313, Revision 2, that exist in its RCPB piping, and discuss the adequacy of the augmented inspection programs in light of the MELLLA+ operation on a plant-specific basis.

12.10 ECCS-LOCA OFF-RATED MULTIPLIER (SECTION 4.3.1.3)

a) The plant-specific application will provide the 10 CFR Part 50, Appendix K, and the nominal PCTs calculated at the rated EPU power/rated CF, rated EPU power/minimum CF, at the low-flow MELLLA+ boundary (Transition Statepoint). For the limiting statepoint, both the upper bound and the licensing PCT will be reported. The M+SAR will justify why the transition statepoint ECCS-LOCA response bounds the 55 percent CF statepoint. The M+SAR will provide discussion on what power/flow combination scoping calculations were performed to identify the limiting statepoints in terms of DBA-LOCA PCT response for the operation within the MELLLA+ boundary. The M+ SAR will justify that the upper bound and licensing basis PCT provided is in fact the limiting PCT considering uncertainty applications to the non-limiting statepoints.

b) LOCA analysis is not performed on cycle-specific basis; therefore, the thermal limits applied in the M+SAR LOCA analysis for the 55 percent CF MELLLA+ statepoint and/or the transition statepoint must be either bounding or consistent with cycle-specific off-rated limits. The COLR and the SRLR will contain confirmation that the off-rated limits assumed in the ECCS-LOCA analyses bound the cycle-specific off-rated limits calculated for the MELLLA+ operation. Every future cycle reload shall confirm that the cycle-specific off-rated thermal limits applied at the 55 percent CF and/or the transition statepoints are consistent with those assumed in the plant-specific ECCS-LOCA analyses.

c) Off-rated limits will not be applied to the minimum CF statepoint.

d) If credit is taken for these off-rated limits, the plant will be required to apply these limits during core monitoring.

12.11 ECCS-LOCA AXIAL POWER DISTRIBUTION EVALUATION (SECTION 4.3.1.4)

For MELLLA+ applications, the small and large break ECCS-LOCA analyses will include top-peaked and mid-peaked power shape in establishing the MAPLHGR and determining the PCT. This limitation is applicable to both the licensing bases PCT and the upper bound PCT. The plant-specific applications will report the limiting small and large break licensing basis and upper bound PCTs.

12.12 ECCS-LOCA REPORTING (SECTION 4.3.1.5)

a) Both the nominal and Appendix K PCTs should be reported for all of the calculated statepoints, and

b) The plant-variable and uncertainties currently applied will be used, unless the NRC staff specifically approves a different plant variable uncertainty method for application to the non-rated statepoints.

12.13 SMALL BREAK LOCA (SECTION 4.3.2.4)

Small break LOCA analysis will be performed at the MELLLA+ minimum CF and the transition statepoints for those plants that: (1) are small break LOCA limited based on small break LOCA analysis performed at the rated EPU conditions; or (2) have margins of less than or equal to [[ ]] relative to the Appendix K or the licensing basis PCT.

12.14 BREAK SPECTRUM (SECTION 4.3.3)

The scope of small break LOCA analysis for MELLLA+ operation relies upon the EPU small break LOCA analysis results. Therefore, the NRC staff concludes that for plants that will implement MELLLA+, sufficient small break sizes should be analyzed at the rated EPU power level to ensure that the peak PCT break size is identified.

12.15 BYPASS VOIDING ABOVE THE D-LEVEL (SECTION 5.1.1.5.3)

Plant-specific MELLLA+ applications shall identify where in the MELLLA+ upper boundary the bypass voiding greater than 5 percent will occur above the D-level. The licensee shall provide in the plant-specific submittal the operator actions and procedures that will mitigate the impact of the bypass voiding on the TIPs and the core simulator used to monitor the fuel performance. The plant-specific submittal shall also provide discussion on what impact the bypass voiding greater than 5 percent will have on the NMS as defined in Section 5.1.1.5. The NRC staff will evaluate on plant-specific bases acceptability of bypass voiding above D level.

12.16 RWE (SECTION 9.1.1.2)

Plants operating at the MELLLA+ operating domain shall perform RWE analyses to confirm the adequacy of the generic RBM setpoints. The M+SAR shall provide a discussion of the analyses performed and the results.

12.17 ATWS LOOP (SECTION 9.3.1.1)

As specified in LTR NEDC-33006P, at least two plant-specific ATWS calculations must be performed: MSIVC and PRFO. In addition, if RHR capability is affected by LOOP, then a third plant-specific ATWS calculation must be performed that includes the reduced RHR capability. To evaluate the effect of reduced RHR capacity during LOOP, the plant-specific ATWS calculation must be performed for a sufficiently large period of time after HSBW injection is complete to guarantee that the suppression pool temperature is cooling, indicating that the RHR capacity is greater than the decay heat generation. The plant-specific application should

include evaluation of the safety system performance during the long-term cooling phase, in terms of available NPSH.

12.18 ATWS TRACG ANALYSIS (SECTION 9.3.1.3)

a) For plants that do not achieve hot shutdown prior to reaching the heat capacity temperature limit (HCTL) based on the licensing ODYN code calculation, plant-specific MELLLA+ implementations must perform best-estimate TRACG calculations on a plant-specific basis. The TRACG analysis will account for all plant parameters, including water-level control strategy and all plant-specific emergency operating procedure (EOP) actions.

b) The TRACG calculation is not required if the plant increases the boron-10 concentration/enrichment so that the integrated heat load to containment calculated by the licensing ODYN calculation does not change with respect to a reference OLTP/75 percent flow ODYN calculation.

c) Peak cladding temperature (PCT) for both phases of the transient (initial overpressure and emergency depressurization) must be evaluated on a plant-specific basis with the TRACG ATWS calculation.

d) In general, the plant-specific application will ensure that operation in the MELLLA+ domain is consistent with the assumptions used in the ATWS analysis, including equipment out of service (e.g., FWHOOS, SLO, SRVs, SLC pumps, and RHR pumps, etc.). If assumptions are not satisfied, operation in MELLLA+ is not allowed. The SRLR will specify the prohibited flexibility options for plant-specific MELLLA+ operation, where applicable. For key input parameters, systems and engineering safety features that are important to simulating the ATWS analysis and are specified in the Technical Specification (TS) (e.g., SLCS parameters, ATWS RPT, etc.), the calculation assumptions must be consistent with the allowed TS values and the allowed plant configuration. If the analyses deviate from the allowed TS configuration for long term equipment out of service (i.e., beyond the TS LCO), the plant-specific application will specify and justify the deviation. In addition, the licensee must ensure that all operability requirements are met (e.g., NPSH) by equipment assumed operable in the calculations.

e) Nominal input parameters can be used in the ATWS analyses provided the uncertainty treatment and selection of the values of these input parameters are consistent with the input methods used in the original GE ATWS analyses in NEDE-24222. Treatment of key input parameters in terms of uncertainties applied or plant-specific TS value used can differ from the original NEDE-24222 approach, provided the manner in which it is used yields more conservative ATWS results.

f) The plant-specific application will include tabulation and discussion of the key input parameters and the associated uncertainty treatment.

12.19 PLANT-SPECIFIC ATWS INSTABILITY (SECTION 9.3.3.1)

Until such time that NRC approves a generic solution for ATWS instability calculations for MELLLA+ operation, each plant-specific MELLLA+ application must provide ATWS instability analysis that satisfies the ATWS acceptance criteria listed in SRP Section 15.8. The plant-specific ATWS instability calculation must: (1) be based on the peak-reactivity exposure conditions, (2) model the plant-specific configuration important to ATWS instability response including mixed core, if applicable, and (3) use the regional-mode nodalization scheme. In order to improve the fidelity of the analyses, the plant-specific calculations should be based on

latest NRC-approved neutronic and thermal-hydraulic codes such as TGBLA06/PANAC11 and TRACG04.

12.20 GENERIC ATWS INSTABILITY (SECTION 9.3.3.1)

Once the generic solution is approved, the plant-specific applications must provide confirmation that the generic instability analyses are relevant and applicable to their plant. Applicability confirmation includes review of any differences in plant design or operation that will result in significantly lower stability margins during ATWS such as:

- turbine bypass capacity,
- fraction of steam-driven feedwater pumps,
- any changes in plant design or operation that will significantly increase core inlet subcooling during ATWS events,
- significant differences in radial and axial power distributions,
- hot-channel power-to-flow ratio,
- fuel design changes beyond GE14.

12.21 INDIVIDUAL PLANT EVALUATION (SECTION 10.5)

Licensees that submit a MELLLA+ application should address the plant-specific risk impacts associated with MELLLA+ implementation, consistent with approved guidance documents (e.g., NEDC-32424P-A, NEDC-32523P-A, and NEDC-33004P-A) and the Matrix 13 of RS-001 and re-address the plant-specific risk impacts consistent with the approved guidance documents that were used in their approved EPU application and Matrix 13 of RS-001. If an EPU and MELLLA+ application come to the NRC in parallel, the expectation is that the EPU submittal will have incorporated the MELLLA+ impacts.

12.22 IASCC (SECTION 10.7)

The applicant is to provide a plant-specific IASCC evaluation when implementing MELLLA+, which includes the components that will exceed the IASCC threshold of $5x10^{20}$ n/cm^2 (E>1MeV), the impact of failure of these components on the integrity of the reactor internals and core support structures under licensing design bases conditions, and the inspections that will be performed on components that exceed the IASCC threshold to ensure timely identification of IASCC, should it occur.

12.23 LIMITATIONS FROM THE ATWS RAI EVALUATIONS (APPENDIX A)

12.23.1 Limitation from Appendix A RAI 4-1

See limitation 12.18.d.

12.23.2 Limitation from Appendix A RAI 4-2

The plant-specific ODYN and TRACG key calculation parameters must be provided to the staff so they can verify that all plant-specific automatic settings are modeled properly.

12.23.3 Limitation from Appendix A RAI 11-4

The ATWS peak pressure response would be dependent upon SRVs upper tolerances assumed in the calculations. For each individual SRV, the tolerances used in the analysis must be consistent with or bound the plant-specific SRV performance. The SRV tolerance test data would be statistically treated using the NRC's historical 95/95 approach or any new NRC-approved statistical treatment method. In the event that current EPU experience base shows propensity for valve drift higher than pre-EPU experience base, the plant-specific transient and ATWS analyses would be based on the higher tolerances or justify the reason why the propensity for the higher drift is not applicable the plant's SRVs.

12.23.4 Limitation from Appendix A RAI 13-1

EPG/SAG parameters must be reviewed for applicability to MELLLA+ operation in a plant-specific basis. The plant-specific MELLLA+ application will include a section that discusses the plant-specific EOPs and confirms that the ATWS calculation is consistent with the operator actions.

12.23.5 Limitation from Appendix A RAI 14-5

The conclusions of this LTR and associated SE are limited to reactors operating with a power density lower than 52.5 MW/MLBM/hr for operation at the minimum allowable CF at 120 percent OLTP. Verification that reactor operation will be maintained below this analysis limit must be performed for all plant-specific applications.

12.23.6 Limitation from Appendix A RAI 14-9

For MELLLA+ applications involving GE fuel types beyond GE14 or other vendor fuels, bounding ATWS Instability analysis will be provided to the staff. Note: this limitation does not apply to special test assemblies.

12.23.7 Limitation from Appendix A RAI 14-10

See limitation 12.23.6.

12.23.8 Limitation from Appendix A RAI 14-11

The plant-specific ATWS calculations must account for all plant- and fuel-design-specific features, such as the debris filters.

12.23.9 Limitation from Appendix A RAI 16-1

Plant-specific applications must review the safety system specifications to ensure that all of the assumptions used for the ATWS SE indeed apply to their plant-specific conditions. The NRC staff review will give special attention to crucial safety systems like HPCI, and physical limitations like NPSH and maximum vessel pressure that RCIC and HPCI can inject. The plant-specific application will include a discussion on the licensing bases of the plant in terms of NPSH and system performance. It will also include NPSH and system performance evaluation for the duration of the event.

12.23.10 Limitation from Appendix A RAI 16-3

Plant-specific applications must ensure that an increase in containment pressure resulting from ATWS events with EPU/MELLLA+ operation does not affect adversely the operation of safety-grade equipment.

12.23.11 Limitation from Appendix A RAI 17-1

The plant-specific applications must justify the use of plant-specific suppression pool temperature limits for the ODYN and TRACG calculations that are higher than the HCTL limit for emergency depressurization.

12.24 LIMITATIONS FROM FUEL DEPENDENT ANALYSES RAI EVALUATIONS (APPENDIX B)

12.24.1 Limitation from Appendix B RAI 3

For EPU/MELLLA+ plant-specific applications that use TRACG or any code that has the capability to model in-channel water rod flow, the supporting analysis will use the actual flow configuration.

12.24.2 Limitation from Appendix B RAI 7

The EPU/MELLLA+ application would provide the exit void fraction of the high-powered bundles in the comparison between the EPU/MELLLA+ and the pre-MELLLA+ conditions.

12.24.3 Limitation from Appendix B RAI 17

See limitation 12.6.

12.24.4 Limitation from Appendix B RAI 30

See limitation 12.18.d.

13.0 CONCLUSION

LTR NEDC-33006P defines the approach and provides the basis for an expansion of the operating range for BWR plants that have uprated power, either with or without a change in the operating pressure. The NRC staff performed a comprehensive review and confirmatory analyses, because the reactor operating conditions and plant response during MELLLA+ operation will be outside the current experience base. Based on the NRC staff review of the LTR, the information provided in the RAI responses, the insights from the NRC staff confirmatory analyses, and given the limitations and conditions of this SE, the NRC staff finds LTR NEDC-33006P acceptable for BWR plants with GE/GNF fuel designs through GE14, analyzed with GE methodologies.

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APPENDIX A

NRC STAFF EVALUATION OF ATWS RAI RESPONSES

TO

SAFETY EVALUATION BY

THE OFFICE OF NUCLEAR REACTOR REGULATION

LICENSING TOPICAL REPORT NEDC-33006P

"GENERAL ELECTRIC BOILING WATER REACTOR MAXIMUM EXTENDED LOAD LINE LIMIT ANALYSIS PLUS"

GENERAL ELECTRIC HITACHI NUCLEAR ENERGY AMERICA, LLC

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APPENDIX A ATWS RAI EVALUATIONS

This appendix provides the NRC staff's evaluation of responses to requests for additional information (RAIs). This appendix only provides the RAI question and evaluation, not the RAI response. The RAI responses can be found in References 20, 29, and 30.

NRC RAI 1: LOOP EVENT DISPOSITION

Discuss how it will be determined [[

]]

Include in the plant-specific applications, a discussion of why the RHR cooling capability does or does not affect the plant's ATWS LOOP event response. The plant-specific MELLLA+ safety analysis report (MSAR) should state the bases for confirming that [[

]].

See Reference 31 RAI I-1.1 for RAI response.

Evaluation of RAI 1:
The NRC staff agrees with the main conclusion: LOOP would result in lower peak pressure, and lower PCT. As described in the RAI response, [[

]]

Therefore, it is not evident that the long term cooling will be better under LOOP if RHR capacity is compromised.

Revision 2 of the MELLLA+ LTR (MLTR) methodology specifies that the LOOP analysis will be performed in addition to [[ ]] events if the RHR capability is confirmed to be limited under the LOOP event. The discussion in the RAI response should be included in the MLTR in order to explain what the confirmation would entail. Section 9.3.1.1 of this SE provides additional discussion and an associated limitation.

NRC RAI 2: INADVERTENT OPENING OF RELIEF VALVE DISPOSITION (IORV)

IORV is a long-term depressurization transient that affects the long-term suppression pool heatup. This event does not result in high peak pressure in the short-term ATWS response. However, since the recirculation pump trip (RPT) and the standby liquid control (SLC) initiation occur later, the amount of energy discharged into the suppression pool in the long term could be high. The plant's response to this event may depend on the RHR cooling capability and the initial operating conditions of the plant. Considering the higher core reactivity for the extended power uprate (EPU)/MELLLA+ condition during an ATWS event and the plant's unchanged RHR cooling capabilities, explain the basis for concluding that the IORV event would not result in a limiting suppression pool temperature during the long-term ATWS recovery period. Justify why this conclusion holds for all of the BWR fleet.

See Reference 31 RAI I-1.2 for RAI response.

Evaluation of RAI 2:
The NRC staff agrees with the conclusion presented in the RAI response. IORV with condenser available will have a lower consequence on suppression pool than isolation ATWS: no significant overpressure, no PCT, lower suppression pool temperature.

NRC RAI 3: DETERMINING PEAK CLADDING TEMPERATURE (PCT)

RAI 3-1: Initial Hot Bundle Operating Conditon

Explain how, during an ATWS event, the hot bundle operation will be constrained by the same operating thermal limits as at the maximum core flow condition. Wouldn't the fuel experience thermal overpower conditions that are higher than the peak design limits?

See References 29 and 31 RAI I-2.1 for RAI response.

Evaluation RAI 3-1:
The RAI response states that in performing the PCT calculations [[

]] In RAI response concludes that conditions assumed make the calculated PCT conservative irrespective of the assumed initial conditions. Therefore, the results from the analysis are bounding and cycle independent.

The NRC staff agrees with RAI response that the ATWS acceptance criteria are:

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

A coolable core geometry is assured by meeting the 2200 °F peak cladding temperature (PCT) and 17% local cladding oxidation acceptance criteria of 10CFR50.46.

The NRC staff accepts the changes in Revision 2 of NEDC-33173P, which proposes that the PCT will be calculated at the M+ 85% statepoint and compared with the CLTP values to show the impact of the change. This calculations is based on the ODYN licensing calculations of the initial pressurization PCT.

The NRC staff expects all BWRs would require depressurization, during an ATWS event under EPU/MELLLA+ conditions. Therefore, there are two significant PCT events: Initial over-pressurization phase, and during the emergency depressurization phase. Depending on the water level strategy followed by the plant, the PCT event for the second phase could be higher than for the first phase. Both PCT phases will be captured by the plant specific TRACG ATWS calculations. ODYN (ATWS licensing code) cannot be used to model the depressurization phase or water level strategy below TAF+5.

RAI 3-2: Determining PCT

Provide a table showing the previous PCT results used to make the assessment. List the MELLLA+ PCT sensitivity analyses the MLTR is referring to. Describe the key assumptions used for the PCT calculations (BWR type, fuel type, rodline and power level, etc.). Identify if ODYN/ISCOR/TASC combination or TRACG was used in calculating the PCT.

See Reference 31 RAI I-2.2 for RAI response.

Evaluation RAI 3-2:
This RAI was based on the initial Revision 1 of NEDC-33006P, which proposed not providing the ATWS PCT calculation.

The ODYN/ISCOR/TASC results were provided. The PCT values are well within the 2200° F limits.

However, the NRC staff expects all BWRs would require depressurization during an ATWS event under EPU/MELLLA+ conditions. Therefore, there are two significant PCT events: Initial over-pressurization phase, and during the emergency depressurization phase. Depending on the water level strategy followed by the plant, the PCT event for the second phase could be higher than for the first phase. Both PCT phases will be captured by the plant specific TRACG ATWS calculations. Revision 2 to the LTR, updated the ATWS section and includes the commitment to provide plant-specific TRACG PCT calculation.

In addition, Table A-1 shows PCT values using ODYN/ISCOR/TASC calculation in which the PCT goes down for the EPU/MELLLA conditions. GEH/GNF-A attributes the lower PCT response for 20% higher power to flow re-distribution because the average channel power increases while the hot channel is assumed to operate the maximum design limit. However, the NRC staff believes that the modeling limitation of the ODYN/ISCOR/TASC code system is contributing to the lower PCT response. The code system does not represent the flatter radial power distribution of the EPU cores, and the distribution of large fraction of high powered bundles in the core. Therefore, the flow redistribution with in the average and the hot bundles modeled in ODYN/ISCOR/TASC may not capture the actual bundle flow distribution of an EPU core, with high batch fraction, more high powered bundles.

Note that the ATWS analysis is performed at the minimum core flow statepoint, which is considered to be more limiting, because the RPT is less effective in adding negative void reactivity relative to ATWS initiated from rated conditions. Therefore, all these calculations are performed at higher rodline than rated OLTP conditions. For example, the MELLLA/OLTP statepoint at 75% core flow statepoint rodline corresponds to higher rodline than the rated OLTP 100% rodline. Therefore, for MELLLA/ 75% core flow and MELLLA/EPU, the reactor may reach similar power levels after RPT. However, as can be seen in Figure 3-2 below, after MSIV closure, the peak neutron flux initiated from MELLLA+ at 85% core is much higher than MELLLA at OLTP/75% core flow statepoint. Therefore, since the initial peak PCT corresponds to the initial power peak due to the MSIVC, it is not clear why the MELLLA+ PCT is lower,

However, with the plant-specific TRACG sensitivity analyses, the core thermal hydraulic conditions will be modeled more accurately.

Table A-1 PCT Comparisons at ELLLA/OLTP, MELLLA/OLTP &105%P, MELLLA/EPU and MELLLA+/EPU

Peak Cladding Temperature (°F)
(All Calculations Based on ODYN/ISCOR/TASC Methodology)

[[					

]

[[

]]

Figure A-1 Neutron Flux response for MSIVC ATWS Event.

NRC RAI 4: ODYN ATWS CALCULATIONS

The NRC staff has reviewed ODYN data for ATWS events for three plants (Plant D, Plant E, and Plant F) at two operating conditions (100 percent OLTP, 75 percent flow, and 120 percent OLTP, 85 percent flow). The following RAIs address the key assumptions and system actuations used for these analyses.

RAI 4-1: Plant D ATWS Response

The Plant D MELLLA+ LTR (NEDC-33063P) states that the peak vessel pressure for an MSIV isolation ATWS is 1457 psig. [[

]]

1. What is the difference between the two calculations?
2. What is the applicable peak pressure limit?
3. If the applicable limit is 1500 psig, is it violated by the ODYN calculation results provided?
4. Give what the peak pressures are for other analyzed ATWS cases, including PRFO.
5. Provide the ODYN results as a function of time for the limiting ATWS event for Brunswick.

See Reference 31 RAI IV-1 for RAI response.

Evaluation RAI 4-1:
The main difference among the various calculations is the number of SRVs assumed out of service. The upgrade to MELLLA+ conditions mandate that all 11 SRVs be operational to satisfy the 1500 psig over-pressure limit for Brunswick. If one SRV is placed out of service, the plant must exit the MELLLA+ region. Table A-2 shows the peak pressure values corresponding to different cycle exposures and with one or no SRVOOS.

Table A-2 Plant D Peak Vessel Pressure at MELLLA+

Event	Cycle Exposure	Number of SRV OOS	Peak Vessel Pressure (psig/sec)
MSIVC	BOC	[[	
MSIVC	EOC		
PRFO	BOC		
PRFO	EOC		
PRFO	EOC		]]

The NRC staff concludes that MELLLA+ operation has a significant effect on the initial over-pressure following isolation events. Note that the TS SRV LCO is attributed to the ASME Overpressure transient analysis. However, the SRVs are relied upon in meeting the ATWS acceptance criteria. As can be seen by this example, the number of SRVOOS affects the plants' capabilities to meet the ATWS acceptance criteria. Therefore, the allowed TS SRVOOS need to be consistent with the number assumed in the ATWS analysis. Section 9.3.1.3 of this SE provides additional discussion and an associated limitation.

RAI 4-2: Timing of Operator Actions

It is customary in safety calculations to allow some time for operator actions. It is apparent from a review of the ODYN results that operator actions occur in very short timeframes. [[

]] Explain the assumptions used for operator actions during these analyses.

See Reference 31 RAI IV-2 for RAI response.

Evaluation RAI 4-2:
For the ODYN calculations, operator actions assume a 2 minute delay. Some ATWS mitigation actions are automatic and, for those, plant-specific settings are used. [[

]]

Limitation:
The plant-specific ODYN and TRACG key calculation parameters must be provided to the staff so they can verify that all plant-specific automatic settings are modeled properly.

RAI 4-3: ODYN Calculation Assumptions

In the Plant D calculation, the water level is raised at exactly 1400 seconds (in the Plant F calculation at exactly 1600 seconds). According to the EPGs, the water level is supposed to be raised when the HSBW has been injected into the core. What is the basis for the exact [[ ]] used? Shouldn't the time when the HSBW is reached be dependent on the SLC injection initiation time?

See Reference 31 RAI IV-3 for RAI response.

Evaluation RAI 4-3:
The water level in the Plant D calculation is [[

]] Therefore, the NRC staff agrees that the ODYN calculations represent the expected timing of operator actions.

RAI 4-4: Limiting ATWS Events and Scenarios

Provide the assumptions used in the ATWS analysis for the EPU/MELLLA+ pilot plant [Plant D] specific calculation (NEDC-33063P). Specifically, what type of ATWS transient is limiting? What are the initial conditions, including the power, flow, and the suppression pool level? What operator actions are assumed? What ATWS mitigation actions are implemented during the transient? What values are used for EOP variables (e.g., HCTL, HSBW, etc.)?

See Reference 31 RAI V-3 for RAI response.

Evaluation RAI 4-4:
Table A-3 shows the requested information, except the EOP variables of HCTL.

Table A-3 Plant D ATWS Analysis Assumptions –

Bounding ATWS events	MSIVC and PRFO
Initial Reactor Power (% OLTP)	120

Initial Core Flow (% NBR Rated)	85
Initial Suppression Pool Mass (lbm)	5,365,000 at minimum water level
Operator Actions Assumed	1. Maintain water level at TAF+5' following feedwater pump trip 2. [[ ]] 3. Increase water level after HSBW injected
ATWS Mitigation Implemented	1. Reduce water level to TAF+5' 2. Inject boron
EOP Variable Used	[[ ]]

NRC RAI 5: IMPACT OF MELLLA+ ON THE HSBW VALUE

Because MELLLA+ operation occurs at a higher control rod line, one would expect the HSBW to increase over the baseline. The analysis assumed the same HSBW value for both MELLLA+ and the previous baseline condition. Under MELLLA+, the HSBW may be higher, leading to a longer time of suppression pool heating before the water level is raised to remix the boron at the bottom of the vessel, which achieves the hot shutdown condition. What is the effect of using a MELLLA+ specific HSBW value on ultimate suppression pool temperature?

See Reference 31 RAI IV-4 for RAI response.

Evaluation RAI 5:
A sensitivity analysis was provided by GEH in the response to this RAI. Table A-4 below provides the sensitivity analysis for all conditions, the ODYN criteria was satisfied; however, using more realistic TRACG analysis, the results show that the suppression pool HCTL will be exceeded and emergency depressurization would be necessary.

Table A-4 Plant D HSBW Sensitivity Results

HSBW (ppm)	HSBW Injection Time (sec)	Level Increase Time (sec)[1]	Approximate Hot Shutdown Time (sec)[2]	Peak Pool Temp (F)
[[				
				]]

[1] The level increase time is the sum of HSBW injection time, SLCS initiation time, and Boron transport time.
[2] [[

]]

NRC RAI 6: SRV INDUCED FLOW OSCILLATIONS

The EPGs instruct the operator that a number of SRVs should be locked open to prevent cycling (and prevent possible mechanical failures). By allowing the SRVs to cycle, the core flow oscillates wildly because of the SRV-induced pressure transients. By increasing the flow values over the non-mixing stagnation flow value in the boron correlation, these wild flow oscillations promote Boron mixing that otherwise would not happen. Explain why it is conservative to allow these wild flow oscillations to continue, thus increasing the amount of boron mixed with the core inlet coolant and reducing the reactor power.

See Reference 31 RAI IV-5 for RAI response.

Evaluation RAI 6:
[[

]] Therefore, even though the ODYN calculation does not accurately represent the SRV open/close cycles expected during the early phase of the ATWS event in the real event, boron mixing is not enhanced by this fluctuations, and the results are representative of real conditions.

NRC RAI 7: SRV TOLERANCES USED

Section 9.3.1 of the Plant D MELLLA+ LTR (NEDC-33063P) states that the MELLLA+ analysis was performed with 10 percent SRV tolerance, rather than the normally assumed 3 percent tolerance. Provide an explanation of the detailed SRV lifting pressures (including the tolerance) and the percent of nameplate flow used for the calculations.

See Reference 31 RAI IV-6 for RAI response.

Evaluation RAI 7:
The Plant D ATWS overpressure evaluation was performed with the lowest opening SRV tolerance of 10%. All other 10 groups use the standard 3% tolerance. However, the RAI response does not discuss whether the 10% tolerance was used to bound potential high valve drifts of the lowest opening SRVs. In addition, the RAI response states that SRV nameplate capacity is 829,000 lbm/hr, based on reference pressure of 1080 psig. Further discussion is required on the SRVs capacity through out the event scenario.

Recent experience indicates that some licensees have used valve tolerances that are less than the actual valve performance, even though LERs had been issued on the results of valve testing data outside Tech Spec ranges. The valve tolerance uncertainty treatment should be based on a NRC approved method or the historically recommended NRC method of 95/95 or plant-specific setpoint methodology needs to be reviewed and approved.

NRC RAI 8: ATWS SEQUENCE OF EVENTS

Provide the sequence of events (including SLC injection and water level reduction times) for these calculations. Specify the actuation setpoints and initiation times. What are they based on?

See Reference 31 RAI IV-7 for RAI response.

Evaluation RAI 8:
The sequence of events for Plant D with for the one SRVOOS was provided. Note that the peak pressure with 1 SRVOOS exceeded the ATWS pressure limit of 1500 psig.

Table A-5 Sequence of Events for an MSIVC Event

Item	Response	M+ Event Time (sec)
1	MSIV Isolation Initiates	[[
2	High Pressure ATWS Setpoint	
3	MSIVs Closed	
4	Peak Neutron Flux	
5	Opening of the First Relief Valve Tripped	
6	Recirculation Pumps Tripped	
7	Peak Heat Flux Occurs	
8	Peak Vessel Pressure	
9	Feedwater Reduction Initiated	
10	BIIT Reached	
11	PCT Occurs	
12	SLCS Pumps Start	
13	Water Level Increased	
14	Hot Shutdown Achieved [[ ]]	
15	Peak Suppression Pool Temperature	]]

Table A-6 Sequence of Events for an PRFO Event

Item	Response-	M+ Event Time (sec)
1	Turbine Control and Bypass Valves Start Open	[[
2	MSIV Closure Initiated by Low Steamline Pressure	
3	MSIVs Fully Closed	
4	Peak Neutron Flux	
5	High Pressure ATWS Setpoint Tripped	
6	Opening of the First Relief Valve Tripped	
7	Recirculation Pumps Tripped	
8	Peak Heat Flux Occurs	
9	Peak Vessel Pressure	
10	Feedwater Pumps Runback Initiated	
11	BIIT Reached	
12	PCT Occurs	
13	SLCS Pumps Start	
14	Water Level Increased	
15	Hot Shutdown Achieved [[ ]]	
16	Peak Suppression Pool Temperature	]]

Table A-7 Initial Valve Opening Timing for an MSIVC Event and a PRFO Event

Valve Group	MSIVC Event	PRFO Event
1	[[	
2		
3		
4		
5		
6		
7		
8		
9		
10		
11		]]

NRC RAI 9: PCT DISPOSITION AND FUTURE FUEL DESIGNS

Justify why the sensitivity results, based on performance of GE fuel (up to GE14), form the bases for [[ ]] Alternatively, state that the coolable geometry (e.g., PCT) and the 17 percent local cladding oxidation acceptance limit for the ATWS analyses would be demonstrated on a plant-specific basis, if another vendor's fuel, new GNF fuel, or mixed vendor cores are involved. In the latter case, revise the MLTR and include a specific applicability statement.

See References 20 and 31 RAI I-2.3 for RAI response.

Evaluation of RAI 9:
This RAI is based on Revision 1 of NEDC-33006P. The NRC staff disagrees with the evaluation. Even though large margin has been demonstrated for the lead plant, the consequences of failure of this criterion are large. Plant specific calculations will guarantee that the criterion is met. See details in section NRC RAI I-3.3 below.

Since licensee's can introduce new fuel without explicit NRC approval, it is not clear what ATWS analysis would be performed during the new fuel introduction phase. The plant-specific application would include TRACG ATWS analysis. In evaluating the TRACG sensitivity analysis, the NRC staff would determine on plant-specific bases, if additional limitation is required. This is especially relevant in the applications showing that the plant is limited in terms of PCT.

Revision 2 of the LTR addresses the NRC staff concerns and no limitation is needed.

NRC RAI 10: LIMITING ATWS RESPONSE AND POWER/FLOW STATEPOINT

Explain why the ATWS analysis performed at the minimum core flow statepoint is more limiting than the analysis performed at the maximum achievable core statepoint for the EPU/MELLLA+ operation.

See References 29 and 31 RAI I-2.4 for RAI response.

Evaluation RAI 10:
EPU experience indicates that BWRs are limited by the ATWS peak pressure. Consequently, the sensitivity if the ATWS pressurization response to the initial flow statepoint is important. In the RAI response, GNF-A states that the PCT and the suppression pool response would be more limiting from the minimum core flow statepoint rather than at the rated core flow statepoint.

The power level during the ATWS event (after flow reduction) is controlled mostly by the operating control rod line. The low-flow initial condition at rated power is in a higher control rod line, so the power after the flow reduction will be larger. Although the control rod line is the dominant effect; other mechanism (like flow redistribution) are in play, could result in second order perturbations. This combined with the assumed radial peaking factor at the low flow condition and the higher void conditions could lead to higher PCT. Similar to the ASME overpressure response, it is feasible that the peak ATWS response could be more limiting at the rated or maximum core flow statepoints, depending on the control rod pattern assumed and the exposure. However, considering that the ATWS RPT is less effective if initiated from the minimum flow statepoint (80%) compared to the rated core flow statepoint and the that the

minimum core flow is at ~ 140 % rodline, the NRC staff accepts that the minimum core flow is the limiting statepoint for the ATWS analysis.

NRC RAI 11: APPLICABILITY OF THE ODYN LICENSING METHODOLOGY TO THE MELLLA+ ATWS

The Emergency Procedure Guidelines (EPGs) require a number of operator actions, and they allow a range of water level control strategies during isolation ATWS events, from 2 feet below the feedwater spargers to the minimum steam cooling water level (MSCWL). However, limitations in the approved ODYN methodology only allows for an ATWS calculation with a minimum water level of top-of-active (TAF+5 ft), and do not allow for accurate modeling of all required operator actions (such as depressurization when the heat capacity temperature limit (HCTL) is reached). The relevant question is whether the approved ODYN ATWS methodology provides conservative results that can be used to evaluate the impact of MELLLA+ operation on ATWS performance.

RAI 11-1: ODYN Limitations

Provide a description of the approved ODYN ATWS methodology and its limiting assumptions (e.g., control level at TAF+5, do not depressurize). Provide a description of the treatment of uncertainties in approved ODYN licensing calculations.

See References 29 and 31 RAI I-3.1 for RAI response.

Evaluation RAI 11-1:
The RAI response provided description of ODYN limitations for ATWS analysis as follows:

1. The downcomer level must remain above the jet pump suction and no prolonged level in the active channel is allowed:
2. The duration of the simulation after the upper plenum subcools should be limited.
3. The mass in the separators should not remain zero and, therefore, the code is restricted to applications where the water level remains at or above the top of active fuel plus 5 feet.
4. The code is not presently qualified to perform stability calculations.
5. No lower plenum voiding is allowed.

The NRC staff agrees with these ODYN limitations. However, the response did not address the following part of the RAI. "Provide a description of the treatment of uncertainties in approved ODYN licensing calculations." Does ODYN cover how valve tolerances/uncertainties are going to be addressed?

In MFN 05-081, GEH provided a revised RAI response addressed the uncertainties and valve tolerances assumed in ODYN. The RAI response refers to Section 5.6 of NEDC-24154P-A, (Supplement 1-Volume 4), which states that the ODYN approach is more conservative than the historical licensing philosophy for ATWS. Historically, for ATWS applications, prior regulatory approval has been granted for best-estimate code application based on the low probability of the event, conservatisms in key inputs and the acceptance criteria. [[

]]

The RAI response states that in overall, there is no additional specific treatment of uncertainties as ODYN was demonstrated to be conservative compared to test data and TRACG, and key inputs are set at conservative values.

The NRC staff accepts that the cited uncertainties treatment of valve tolerance and the specific key inputs are conservative, although nominal inputs are used in most parameters. This is acceptable for ATWS.

RAI 11-2: ODYN and TRACG Boron- Mixing Correlation

Provide the exact numerical values of the boron-mixing correlation used by TRACG and ODYN for ATWS calculations and their basis.

See Reference 31 RAI I-3.2 for RAI response.

Evaluation RAI 11-2:
TRACG uses the new mixing correlation developed at UC Santa Barbara in 1995. The values for the UC Santa Barbara correlation used in TRACG were confirmed to the NRC staff by Dr. Theofanous. ODYN uses the more conservative values from the old Vallecitos boron-mixing tests. The correlations used by the two codes are inconsistent, making performance comparison between the codes almost meaningless. However the correlation used by ODYN is the more conservative of the two as it bounds the results of the 1/6 scale tests at Vallecitos and the full-scale tests at Santa Barbara. Therefore, from the point of view of boron mixing correlation ODYN results should be more conservative than the best-estimate TRACG results. The correlation values use for both TRACG and ODYN are shown in the following tables.

Table A-8 Boron mixing correlations for lower plenum injection used by TRACG and ODYN

[[			
			]]

Table A-9 Boron re-mixing correlation used by TRACG and ODYN

[[			
			]]

RAI 11-3: ODYN limitations

What are the remaining limitations of the ODYN ATWS calculations (e.g., ATWS/stability)? How will those limitations be addressed (e.g., use of TRACG for ATWS/stability)?

See Reference 31 RAI I-3.5 for RAI response.

Evaluation of RAI 11-3:
The NRC staff agrees with GEH's evaluation of the limitations of the ODYN code for documentation purpose. ODYN calculations are limited by the following limitations:

1. The downcomer level must remain above the jet pump suction and no prolonged level in the active channel is allowed;
2. The duration of the simulation after the upper plenum subcools should be limited.
3. The mass in the separators should not remain zero and, therefore, the code is restricted to applications where the water level remains at or above the top of active fuel plus 5 feet;
4. The code is not presently qualified to perform stability calculations;
5. No lower plenum voiding is allowed.

Items 1 and 3 limit the ODYN level control strategy at TAF +5'. Item 5 prevents the simulation of depressurization with ODYN code.

The NRC staff agrees with GEH's evaluation that ODYN cannot perform the ATWS stability calculation. The ATWS stability calculation is performed with TRACG code as specified in the NRC approved NEDO-32047-A, "ATWS Rule Issues Relative to BWR Core Thermal-Hydraulic Stability."

RAI 11-4: Containment Performance During Isolation ATWS at MELLLA+ Conditions

Provide a comparison of ODYN results of isolation ATWS simulations at MELLLA+ and original licensed thermal power (OLTP).

See Reference 31 RAI I-6.1 for RAI response.

Evaluation of RAI 11-4:
Table A-10 below presents the information requested. The ODYN calculations indicate that containment over-pressure during ATWS is affected significantly by MELLLA+ operation. During the early part of the isolation transient, the peak vessel pressure increases by as much as 150 psi for some plants, because of limitations on maximum SRV flow. As shown in ATWS RAI 8 response, for a BWR4, the peak pressure reaches a value larger than the allowed 1500 psi assuming previously allowed one SRVOOS. The solution proposed in this case is to eliminate the flexibility to operate with one SRV out of service. If all SRVs are assumed operational for the specific BWR/4s analyzed, the calculated peak vessel pressure is less than the 1500 psi limit.

We conclude that MELLLA+ operation has a significant detrimental effect on the peak vessel pressure following an isolation ATWS. Because the margin to allowed peak pressure is so small, some plants may not be able to operate at MELLLA+ conditions. Under these circumstances, the licensees tend to re-evaluate their calculation assumptions and perform a new calculation with these new assumptions that satisfies the limits. The NRC staff must review these changes in assumptions to guarantee that they are within their allowed technical specifications. For example, it is acceptable to change Tech Specs to require all SRVs to be

operable and then take credit for it. It is not acceptable to perform a probabilistic analysis and conclude that SRVs are available most of the time; therefore we can assume they are operable even though Tech Specs allows the plant to operate with a SRV out of service.

Limitation:
The ATWS peak pressure response would be dependent upon SRVs upper tolerances assumed in the calculations. For each individual SRV, the tolerances used in the analysis must be consistent with or bound the plant-specific SRV performance. The SRV tolerance test data would be statistically treated using the NRC's historical 95/95 approach or any new NRC-approved statistical treatment method. In the event that current EPU experience base shows propensity for valve drift higher than pre-EPU experience base, the plant-specific transient and ATWS analyses would be based on the higher tolerances or justify the reason why the propensity for the higher drift is not applicable the plant's SRVs.

Table A-10 OLTP and MELLLA+ ATWS Results Comparisons

[[		
		]]

RAI 11-5: Actuation Sequence

For the above cases, provide the sequence of events (system and equipment actuation and operator actions for the mitigated cases) and the corresponding times. For example, for the MSIVC mitigated case, tabulate when the high pressure ATWS setpoint is reached, main steam isolation valve (MSIV) closes, ATWS-RPT occurs, peak vessel pressure is reached, feedwater(FW) reduction is initiated, boron injection initiation temperature (BIIT) is reached, SLC pumps starts, and water level increases.

See Reference 31 RAI I-6.2 for RAI response.

Evaluation of RAI 11-5:

The table below shows the system and component actuation setpoints for ATWS initiated from the MELLLA and MELLLA+ minimum flow statepoints. Note that the MELLLA and MELLLA+ minimum flow statepoints correspond to the approximately 120% and 140% rodlines as oppose to rated OLTP which corresponds to the 100% rodline.

Table A-11 BWR/4 (standpipe injection)

Item	Response	OLTP Time (sec)	M+ Event Time (sec)
1	[[		
2			
3			
4			
5			
6			
7			
8			
9			
10			
11			
12			
13			
14			
15			]]

Table A-12 BWR/6 (HPCS boron spray)

Item	Response	OLTP Time (sec)	M+ Event Time (sec)
1	[[		
2			
3			
4			
5			
6			
7			
8			
9			
10			

11			
12			
13			
14			]]

Note:
(1) For upper plenum boron injection plants, the water level stays at TAF or TAF+5' during the ATWS event. The operators do not need to raise water to promote boron mixing because the boron stratification is not an issue.

There appears to be inconsistencies in hot shutdown times for MELLLA and MELLLA+ operation calculations in which the reasoning is not clear. For the BWR/4, the MELLLA+ hot shutdown is achieved earlier ([[ ]]). The peak suppression pool occurs later ([[ ]]). It is not clear whether the hot shutdown is reached earlier because of higher Boron-10 concentration. Similarly, it is not clear why the peak suppression pool temperature occurs earlier for the BWR/6 ([[ ]]).

[[ ]] The RAI response (MFN 05-081) states [[ ]] The RAI response justifies the differences in the hot shutdowns times, [[ ]] the use of this criteria can cause variations in the reported time to hot shutdown.

The key inputs and assumptions used in the ATWS analyses are flagged as audit items. In plant-specific application review, the NRC staff is recommended to audit the assumptions and key inputs used in the ATWS analysis.

RAI 11-6: ATWS Response at OLTP, 5% Stretch, EPU and MELLLA+ Comparison

For all BWRs, tabulate the ATWS results (e.g., peak pressure, suppression pool temperature) before the 5 percent power stretch (if available), after the 5 percent power stretch (if applicable), and after EPU and EPU/MELLLA+. Include in the table the results from the initial GEH generic ATWS analyses. Since the initial plant licensing, many BWRs have adopted range-of operating condition changes that affect their ATWS response. These changes include increases in the fuel cycle length (cycle extension from 18 months to 24 months), power (from 5 percent to 20 percent uprates above the original licensed thermal power), and licensed operating domain (LLLL, ELLLA, MELLLA, maximum core flow). The objective of this table is to assess how the previous changes in the operating conditions affected BWR plants' ATWS margins. This would also serve as a means to evaluate the capability of BWRs to meet the vessel and containment response with the additional EPU/MELLLA+ changes. The NRC staff acknowledges that GEH may not have access to the plant-specific ATWS analysis-of-record for plants with other reload vendors.

See Reference 31 RAI I-6.3 for RAI response.

Evaluation of RAI 11-6:
Note that this RAI pre-dates the agreed upon resolution that TRACG sensitivity analysis will be performed.

The data provided indicate that peak vessel pressure is significantly affected by EPU/MELLLA+ operation (See Section I.6.1above). The suppression pool temperature calculated by the ODYN procedure does not appear to be affected as significantly. This is most likely caused by the extremely conservative nature of the ODYN calculation. However, the fact that the HCTL limit is reached and emergency depressurization will be required, makes it hard to compare suppression pool temperature calculations without depressurization.

Tables A-13 and A-14 present the peak vessel pressure and suppression pool data. Due to the historical ATWS code transition from REDY to ODYN, the results are not based on consistent methodology.

Table A-13 Vessel Pressure (psig) Database

[[				

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Table A-14 Suppression Pool Temperature (°F) Database

[[				
				]]

Note that Table A-14 shows that the suppression pool temperature reaches HCTL, even for operation at the MELLLA operating domain.

NRC RAI 12: IMPACT OF DEPRESSURIZATION ON CONTAINMENT AND CORE INTEGRITY

When following the EPGs, operators are required to depressurize the reactor if the HCTL is reached during the transient. The approved ODYN licensing methodology does not reflect this operator action (the suppression pool continues to heat up after HCTL is reached and the depressurization is ignored). Even though the ODYN licensing methodology may be conservative, ODYN results cannot determine whether the reactor fuel reaches PCT limits that may affect long-term coolability. Thus, TRACG calculations are required to evaluate the impact on fuel PCT limits of depressurization.

RAI 12-1: ODYN/TRACG Comparison

Provide the results of a set of TRACG calculations to evaluate the effect of the ODYN modeling limitations. Compare the TRACG results to the ODYN licensing calculation, including the PCTs. At a minimum, provide TRACG calculations based on limiting conditions that follow the EPGs (i.e., depressurization if HCTL is reached) at the three water level setpoints: TAF+5, TAF, and MSCWL and compare to the ODYN licensing methodology results.

See References 20 and 31 RAI I-3.3 for RAI response.

Evaluation of RAI 12-1:
TRACG sensitivity calculations with water level strategy at TAF+5, TAF, and TAF-2 were provided. These TRACG calculations followed the Emergency Operating Guidelines and core emergency depressurization was required at approximately 600 seconds into the transient, well before the Hot Shutdown Boron Weight (HSBW) was injected. Figure A-2 shows the integrated SRV flow for the three TRACG calculations and one ODYN calculation. The ODYN integrated SRV flow bounds the TRACG results for all three strategies. Note: these are TRACG calculations where the SRVs are assumed to remain open for the whole transient (after the 600 sec emergency depressurization is initiated). Other TRACG calculations were performed later where the SRVs are re-closed once 50 psig is reached per EOPs. Note that re-closing the SRVs results in a pressure perturbation that could induces re-criticality and the reactor power increases. Plant-specific application will provide the TRACG sensitivity analyses. The specific data provided in Figure A-2 indicates ODYN SRV flow is conservative in the latter phase of the event. Staff confirmatory analysis based on ODYN and TRACG SRV data used in CONTAIN do not show high level of differences in the ultimate suppression pool temperatures (See Figure 9-8 of the SE).

[[

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Figure A-2 Comparison of integrated SRV flow calculated by ODYN and TRACG at several water level strategies with emergency depressurization

[[

]]

Figure A-3 Comparison of peak pressure calculated by ODYN and TRACG at several water level strategies

Figure A-3 shows a comparison of the peak pressure calculated by TRACG and ODYN. ODYN has already been licensed for peak pressure calculations (by NEDE-24154-P-A, Licensing Topical Report, Qualification of the One-Dimensional Core Transient Model For Boiling Water Reactors Volume 3, Application of One-Dimensional Transient Model to Licensing Basis Transients August 1986). The NRC staff agrees with the conclusion that ODYN peak pressure calculations are conservative with respect to TRACG.

The ODYN PCT results were not presented by GEH as response to this RAI. Figure A-4 shows the PCT values calculated by TRACG for an isolation ATWS following the EOP procedures. This figure shows two significant temperature excursions: the initial excursions (~10 sec) which is related to the initial over pressurization transient, and the second excursion (~1000 sec) which is caused by a core dryout condition during the emergency depressurization. Note that the core dryout condition is more severe for the low water level control strategies (TAF-2) because the initial core coolant inventory is smaller when the emergency depressurization initiates.

[[

]]

Figure A-4 Peak fuel clad temperature calculated by TRACG for isolation ATWS. During emergency de-pressurization, the core uncovers and clad over-heating occurs, but clad temperature criteria are satisfied

The NRC staff disagrees with the generic disposition of the PCT issue. The plant-specific ODYN calculations cannot predict the depressurization dryout, but it can estimate the initial PCT excursion on a plant-specific basis. The resolution of this issue is documented Revision 2 of LTR.

RAI 12-2: Key Parameters during Depressurization Scenario

Based on the data provided above, demonstrate whether the approved ODYN ATWS methodology is conservative relative to TRACG analyses following the emergency operating procedures (EOPs). Compare the results of the ODYN and TRACG (at different water levels) in terms of meeting the ATWS acceptance criteria. Demonstrate that: (1) the TRACG sensitivity analyses and results are bounding or conservative for all the BWR fleet for EPU/MELLLA+ operating conditions, or (2) that the plant-specific ODYN analyses based on the TAF+5 water level strategy would bound the TRACG sensitivity analyses for all of the BWR fleet, or (3) propose a margin criteria for the ATWS acceptance criteria such that a TRACG analyses following the EOP would be performed for the plant-specific application if the margin criteria is not met. Provide at least the following parameters for the three water level strategies, core and vessel void fractions, fuel temperature profiles and time evolution, boron concentrations at several elevations in the lower plenum, recirculation flow, pressure, power levels, bypass voiding with in the vessel (lower and upper levels) and core wide reactivity.

See References 20 and 31 RAI I-3.4 for RAI response.

Evaluation of RAI 12-2:
The NRC staff agrees that the proposed plant-specific ODYN ATWS calculations bound the best-estimate TRACG results for:

1. Peak pressure calculations, and
2. PCT for the first temperature peak induced by the overpressure transient.

However, the NRC staff concludes that plant-specific ODYN ATWS calculations do not bound the best-estimate TRACG results for PCT temperature for the second temperature peak caused by core dryout during depressurization.

In addition, the ODYN ATWS calculations have both conservative and non-conservative assumptions, and bear little resemblance to the real ATWS transient. Comparison with sample TRACG best estimate ATWS calculations appears to indicate that the conservative assumptions compensate the non-conservative assumption and the overall ODYN result is likely to be conservative (i.e., yield a larger suppression pool temperature.) However, the NRC staff finds it hard to justify the use of the low-fidelity ODYN calculations to guarantee that the ATWS criteria are satisfied on a plant-specific basis. Therefore, the NRC staff does not concur with the proposed ODYN-based methodology for plant-specific ATWS analysis and recommends that best-estimate TRACG calculations be performed to confirm peak pressure.

At the request of the NRC staff, GE re-ran TRACG allowing the SRVs to reclose once 50 psig is reached per EOPs. Re-closing the SRVs results in a pressure perturbation that induces re-criticality and the reactor power increases. The results of these new calculations are shown as Figures 9-2 and 9-3 of the SE. The re-criticality periods are apparent in Figure 9-2 of the SE. They appear to be random in nature, amplitude and duration. Most have relatively low power levels (of the order of 20 to 30%), but some power spikes with power >100% are observed. Figure 9-3 of the SE shows that the reactor pressure during re-criticality periods is as high as 2 Mpa (300 psi), and it has some random characteristics. Figures 9-7 through 9-10 in this MELLLA+ SER show the integrated SRV flow and PCT, indicating that the re-criticality periods around 1500 seconds have a small effect on the overall suppression pool heat load or PCT. Note the large PCT transient in Fig 10 is during the de-pressurization stage and is caused by core dryout. The depressurization PCT response is expected to be more severe for the TRACG TAF-2 case, which is not currently available.

The ODYN calculation that was proposed in the original LTR uses a number of non-best-estimate assumptions that are mostly conservative. However, not all of these assumptions are conservative or even representative of real plant operation. For example, the ODYN calculation does not follow the EOP requirement to depressurize the reactor if the Heat Capacity Temperature Limit (HCTL) is reached - this is a non-conservative assumption. Nevertheless, we agree that the long-term integrated SRV flow calculated by the conservative ODYN procedure bounds the results of all the TRACG analysis performed. Since the ODYN calculation is plant specific, the results provide some value by having the actual plant parameters for suppression pool volume, SRV capacity, RHR performance.

However, the NRC staff concludes that ODYN does not model or estimate the physical phenomena that occurs during the depressurization phase. The reactor does not remain in a hot shutdown condition after the emergency de-pressurization. As can be seen from the total core reactivity plot, the reactor becomes critical and continues to be close to critical conditions after the hot shutdown boron weight is injected. The best-estimate TRACG calculations indicate some random behavior and large sensitivity to plant-specific assumptions, such as the specific ATWS water level control strategy (e.g., TAF+5' versus TAF-2') specified in the plant

EOPs. When re-criticality occurs, occasionally large power spikes (>100% nominal) are observed in some of the available runs (TAF+5 and TAF). In all cases, the reactor re-pressurizes to an intermediate pressure and remains at significant power for minutes and several cycles of de- and re-pressurization are observed for several minutes. The impact of the recriticality on the PCT and the long term cooling can only be determined by performing plant-specific calculations. For this reason, a limitation has been placed to perform best-estimate TRACG calculation on plant-specific basis. Section 9.3.1.3 of this SE provides additional discussion and an associated limitation.

RAI 12-3: Stages and Timing of the depressurization

Describe the stages and timing of the depressurization event that was modeled. Is boron mixing enhanced by this event using TRACG as opposed to the ODYN licensing methodology?

See Reference 31 RAI I-5.2 for RAI response.

Evaluation of RAI 12-3:
The data was provided. Boron mixing is enhanced by the depressurization, but the relative enhancement depends on the particular scenario. Specifically, in this simulations, the boron mixing was enhanced for the low water level scenarios (TAF-2 and TAF), but not for the high water level (TAF+5).

RAI 12-4: Core Void Fraction at Decay Heat Levels

Provide a series of steady-state sensitivity analyses to demonstrate that the core will remain subcritical following depressurization. Provide the core void fraction at decay heat levels and approximately 100 psi pressure for a range of core flows (e.g., 5 percent to 15 percent core flow) that could be possible depending on the water level control strategy.

See Reference 31 RAI I-5.3 for RAI response.

Evaluation of RAI 12-4:
The data was provided. ISCOR steady state calculations indicate that the average core void fraction at decay heat power level, 5% core flow, and 100 psi is approximately [[ ]]

which is sufficient to maintain the reactor subcritical if sufficient boron has already been diluted in the core. However, this void fraction can be reduced significantly if the recirculation core flow is increased, so re-criticality is not impossible. Indeed, TRACG and TRACE calculations both show the possibility of re-criticality occurring for most conditions.

NRC RAI 13: EMERGENCY AND ABNORMAL OPERATING PROCEDURES

RAI 13-1: Emergency and Abnormal Operating Procedures Affected by MELLLA+

Provide some specific examples where the EOPs would be affected by MELLLA+ operation. For example, a cursory review of the EPG/severe accident guidelines (SAGs) are examples of areas that need further evaluation and update for determining limiting values. Other variables not mentioned here may be affected.

1. Maximum Pressure for Heat Capacity Temperature Limit Plot (Section 17.5). Section 17.5 defines the procedure for calculation of the HCTL. In the example plots (Figures B- 17-5 and B-17-6), a maximum pressure of 1100 psig is used. However, TRACG calculations show that the pressure during MSIV ATWS is consistently above 1100 psig. Should the

EPG/SAGs be modified for EPU/MELLLA+ operation to require calculation of the HCTL at the expected higher pressures?

2. Hot Shutdown Boron Weight (HSBW) (Section 17.6). The first assumption is that the reactor is operating on the maximum extended operating domain. Clearly this assumption should be changed to the corner of the MELLLA+ domain. Assumption #6 specifies an operating pressure of 1100 psia. However, TRACG calculations show that during ATWS from EPU/MELLLA+ the expected pressures are significantly higher than 1100 psia.
3. Boron Injection Initiation Temperature. The BIIT is defined as the suppression pool temperature that will allow for injection of the HSBW without reaching the suppression pool HCTL. Should the BIIT curve be modified under MELLLA+ operation?
4. Minimum Number of Safety Relief Valves (SRVs) Required for Decay Heat Removal (Section 17.21). With EPU/MELLLA+, the expected decay heat levels should be higher. Will the minimum number of SRVs change? Will this number affect any other variables?
5. Minimum Number of SRVs Required for Emergency Depressurization (Section 17.22). With EPU/MELLLA+, the expected ATWS power levels should be higher. Will the minimum number of SRVs change? Will this number affect any other variables?
6. Minimum Steam Cooling Pressure (Section 17.23). With EPU/MELLLA+, the expected ATWS power levels should be higher. Will the minimum steam cooling pressure be higher? If the pressure is higher, will this affect any other variables?
7. Minimum Steam Cooling Reactor Pressure Vessel (RPV) Water Level (Section 17.24). With EPU/MELLLA+, the expected ATWS power levels should be higher. Will the minimum steam cooling RPV water level change? If the level does change, how does it affect any other variables?
8. Minimum Zero-Injection RPV Water Level (Section 17.25). With EPU/MELLLA+, the expected ATWS power levels should be higher. Will the minimum zero-injection RPV water level change? If the level does change, how does this affect any other variables?

See Reference 31 RAI I-7.1 for RAI response.

Evaluation of RAI 13-1:
The NRC staff agrees with GEH's evaluation and includes a limitation requiring the review of the EPG/SAG parameters in a plant specific basis.

Limitation:

EPG/SAG parameters must be reviewed for applicability to MELLLA+ operation in a plant-specific basis. The plant-specific MELLLA+ application will include a section that discusses the plant-specific EOPs and confirms that the ATWS calculation is consistent with the operator actions.

RAI 13-2: EOP Guidance

Since most of these parameters are likely to be affected by MELLLA+ operation in all plants, provide the justification why the LTR does not provide generic guidance on these parameters.

See Reference 31 RAI I-7.2 for RAI response.

Evaluation of RAI 13-2:
GEH states that The LTR does not provide generic guidance because the BWROG is the owning body for the EPG/SAG, including the technical bases descriptions, and they have already completed a generic evaluation of the EOP curves and limits that are potentially

affected by changes to reactor power and operating domain. This is fully sufficient to ensure that plant EOPs are updated appropriately for MELLLA+ implementation.

The NRC staff agrees with GEH's position and, therefore, includes a limitation requiring the review of the EPG/SAG parameters in a plant specific basis

NRC RAI 14: ATWS/STABILITY ANALYSES

A major concern for the nonisolation turbine trip ATWS is the presence and impact of unstable large power oscillations, which occur when the flow is reduced and the feedwater temperature cools down as a result of the turbine trip. To manage the consequences of these large power oscillations, the EPGs prescribe a number of mitigation actions intended primarily to suppress these oscillations, including reduction of water level below the feedwater sparger and early boron injection. MELLLA+ operation increases the operating control rod line and increases the likelihood and the resulting amplitude of large power oscillations during ATWS events. The relevant question is whether the EPG mitigation actions are still effective under MELLLA+ conditions.

RAI 14-1: Non-Isolation ATWS with Mitigation

Provide the results of a TRACG calculation for a non-isolation ATWS with the prescribed mitigation actions. Compare to the TRACG results without mitigation actions. Provide the fraction of the core that reaches PCT limits during the non-isolation ATWS with and without mitigation actions.

See Reference 31 RAI I-4.1 for RAI response.

Evaluation of RAI 14-1:
The requested calculation was provided by GEH. Since the transient analyzed is a non-isolation ATWS, the condenser is available and the suppression pool temperature does not change. .The main concern here is whether the mitigation actions prescribed by the Emergency Operating Guidelines are still effective to mitigate the instability when the reactor initial condition is within the MELLLA+ domain.

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Figure A-5 Average core power and flow following a non-isolation turbine trip with bypass ATWS. (No EOPs stability mitigation actions followed)

Figure A-5 shows the average core power and flow following a non-isolation turbine trip with bypass ATWS where the EOP-mandated mitigation actions were not followed. This transient was determined to be the bounding case for oscillation amplitude because the turbine trip cuts off the supply of steam to the feedwater heaters, causing the feedwater temperature to drop to condensate storage tank temperature. This causes a large increase in core inlet subcooling, which results in a very significant increase in power level. At the resulting power levels, the instability grows to a large amplitude limit cycle.

This scenario was analyzed in 1992 and reported in NEDO-32164, and the consequences were found unacceptable by both the industry and the NRC staff. As a result the Emergency Procedure Guidelines (EPGs) were updated to include stability mitigation actions. These mitigation actions include: early boron injection and early water level reduction to below the feedwater spargers to preheat the incoming cold feedwater with vessel steam.

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Figure A-6 Average core power and flow following a non-isolation turbine trip with bypass ATWS. (Performed following EOPs water level is lowered below the FW spargers)

The key issue is whether the mitigation actions studied in NEDO-32164 and implemented in EOPs are still effective under MELLLA+ operating domain. To this end, GEH performed a TRACG calculation following the EOP mitigation actions. The results are shown in Figure A-6. For this transient the water level is lowered to TAF at time 120 seconds, to simulate a typical operator action delay. As seen in this figure, the water level reduction and early boron injection mitigate the unstable oscillations in approximately 80 seconds after operator action is initiated.

The EOPs require that the water level be lowered to at least 2 ft below the sparger. When the cold feedwater enters the downcomer, a small fraction wets the vessel surfaces and runs down along them. The remaining water is dispersed in the downcomer steam environment and exchanges enthalpy rapidly with the steam phase. Thus, lowering the water level has two effects: (1) it preheats the feedwater before it enters the core, thus reducing the effective core reactivity and thermal power generation. (2) It condenses dome steam, thus reducing the steam line flow and the heat load to the suppression pool. Lowering the water level is the most effective prompt-effect action that can be taken during ATWS.

Figures A-7 and A-8 present the limiting bundle powers and PCT. As can be seen from these plots limited fuel damage is expected. Since for the EPU/MELLLA+ operating strategies, number of maximum powered bundles increase, the % of core experiencing fuel damage is expected to increase. Figures A-9 and A-10 provide the core and vessel parameters through the course of the scenario, showing the changes in the inlet subcooling, vessel water level and the FW temperature and flow. As modeled, the FW temperature decreases as the preheating is lost

and the FW temperature equilibrates to the condenser temperature. Figure A-11 shows the vessel pressure.

For non-isolation ATWS with instability, the NRC staff concluded that:

1. Operation in MELLLA+ is detrimental to ATWS/Stability, because it increases the effective rod line following the recirculation pump trip; thus increasing the ATWS power level, the probability on instabilities during ATWS, and the probability that the unstable oscillations will grow to very large amplitudes
2. However, the ATWS/Stability mitigation actions prescribed in the EPGs are still effective even when operating the reactor in the MELLLA+ domain. Thus, large amplitude unstable oscillations will be mitigated in relatively short time by operator actions prescribed in the EPGs.

[[

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Figure A-7 Limiting Bundle Power (Based on EOP Actions)

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Figure A-8 Limiting Bundle PCTs (Based on EOP Actions)

[[

]]

Figure A-9 Core And Vessel Parameters (Based on EOP Actions)

[[

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Figure A-10 FW Temperature and Core Inlet Subcooling (Based on EOP Actions)

[[

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Figure A-11 Core Pressure (Based on EOP Actions)

RAI 14-2: Power and PCT for Implicit and Explicit Numeric

Provide the results of a TRACG calculation for a full-isolation ATWS with depressurization using the TRACG stability numerics.

See Reference 31 RAI I-4.2 for RAI response.

Evaluation of RAI 14-2:
The core power and PCT responses were provided for both, the implicit numeric scheme and the explicit (stability) numerics. Figures A-11 and A-12 show that the differences are not significant. Note that the analysis does not include depressurization.

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Figure A-12 Reactor Power

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Figure A-13 PCT

RAI 14-3: Effectiveness of the Mitigation Actions

Are the mitigation actions prescribed by the EPGs effective to manage ATWS/Stability concerns under MELLLA+ operating conditions?

See Reference 31 RAI I-4.3 for RAI response.

Evaluation RAI 14-3:

Based on the results of the above analysis, the NRC staff agrees with the conclusion that the operator actions prescribed in the emergency procedure guidelines (EPGs) are effective in managing ATWS instability concerns under MELLLA+ operating conditions. It is the responsibility of the fuel vendors and licensees to ensure that this conclusion remains applicable and valid for future changes of fuels design and/or operating strategy. The basis for the current analysis can be found in the RAI responses and the topical report.

RAI 14-4: Documentation

Table 9-5 of the MLTR lists the fuel response for the set of ATWS instability analyses. Figures 9-5 to 9-11 of the MLTR show the fuel response for the high-powered bundles. For clarity, add sub-titles or footnotes to the figures that identify the statepoints and the initial power to flow conditions. Otherwise, label Table 9-5 and the corresponding figures by case numbers. Expand Table 9-5 to include event type (turbine trip or MSIVC) and the mitigated cases. Footnote the mitigation strategy used.

See Reference 31 RAI II-1.1 for RAI response.

Evaluation of RAI 14-4:
The revised LTR would contain the requested information.

RAI 14-5: Documentation of the Analyzed Conditions

Footnote 2 to Table 9-5 of the MLTR states: [[

]] Please, explain this statement. [[

]]

See Reference 31 RAI II-1.2 for RAI response.

Evaluation RAI 14-5:
GEH's evaluation of ATWS/Stability was performed for [[ ]] which appears to bound all the reactors in the fleet. The NRC staff agrees that this evaluation is sufficiently conservative for all other operating conditions with a smaller initial power density. Other initial conditions (e.g. 120% OLTP and 100%, or lower power-density reactors) are likely to exhibit a less severe stability response during ATWS events.

Limitation:

The conclusions of this LTR and associated SE are limited to reactors operating with a power density lower than [[ ]] for operation at the minimum allowable CF at 120 percent OLTP. Verification that reactor operation will be maintained below this analysis limit must be performed for all plant-specific applications.

RAI 14-6: Percent Fuel Failure Rate

Since for EPU/MELLLA+ core design, the number of high-powered bundles will increase, provide an estimate of the percent of the core that may experience PCT greater than 2200° F for the unmitigated cases. Compare this with the conclusions reached from the original ATWS instability evaluations in Reference 14 of the MLTR.

See Reference 31 RAI II-1.3 for RAI response.

Evaluation RAI 14-6:
For the present MELLLA+ evaluation, [[

]] For comparison, 12 percent of the bundles exceeded the limit in the original ATWS instability evaluations in Reference 14. The NRC staff notes that this difference is not statistically significant, because the fuel failures occur during a single unstable pulse (i.e., a power excursion of < 1 sec). During large-amplitude instability events, the pulse amplitudes are essentially random in nature, and the percentage of fuel that exceeds the limit is also random. Considering that the analyses results are based on different analysis conditions and model cores loaded with different fuel designs, the usefulness of the comparisons are limited.

RAI 14-7: Conservatism of the [[ ]]

Considering the variation that exists through the BWR fleet, explain why the [[ ]] is considered to be reasonably bounding.

See Reference 31 RAI II-1.5 for RAI response.

Evaluation of RAI 14-7:
GEH states that a power to flow ratio of [[ ]] is higher than any operating BWRs. The NRC staff has placed a limitation of [[ ]] for the applicability of these analyses. This limitation must be verified for all plant-specific applications.

RAI 14-8: Limiting Plant Configuration

Discuss the scoping criteria, if any, used to select the combination of limiting BWR plant physical configuration characteristics and operating parameters. Explain why [[ ]] was selected for performing the ATWS instability analyses. Include in the discussion the bases for selecting [[ ]] in terms of bypass, FW capacity and type, SRV capacity, and fuel support orifice size. Explain how the limiting power distribution (radial and axial), core loading pattern and core exposures, and the initial minimum critical power ratio were selected in order to analyze the bounding ATWS instability cases for the MELLLA+ operation.

See Reference 31 RAI II-1.2 for RAI response.

Evaluation of RAI 14-8:
ATWS instability events are evaluated to assure that the core coolable geometry criterion is met. These evaluations indicate that ATWS/Stability events from EPU/MELLLA+ conditions clearly violate the criterion if mitigation actions are not employed. Note that this conclusion is not limited to EPU/MELLLA+. ATWS/Stability events without mitigation actions also violate fuel limits when operating at OLTP conditions. The key question is whether the mitigation actions are still effective under EPU/MELLLA+ conditions.

The NRC staff agrees that the use of the [[ ]] TRACG deck is reasonably conservative to perform the above evaluation. This deck includes [[

]] All these characteristics tend to increase the amplitude of the resulting unstable power oscillations.

RAI 14-9: Conservatism of the Use of GE14 Fuel

[[

]] Compare the instability response of the different GE fuel product line.

See Reference 31 RAI II-1.7 for RAI response.

Evaluation of RAI 14-9:
GEH states that the intended fuel for MELLLA+ application is GE14. To address transition cores, [[

]] The NRC staff concurs with GEH's evaluation that, with the mitigation strategies recommended by the EOPs, the core coolable geometry criterion would not be compromised following an ATWS instability event assuming any current GE fuel designs up to GE14.

Limitation:

For MELLLA+ applications involving GE fuel types beyond GE14 or other vendor fuels, bounding ATWS Instability analysis will be provided to the staff. Note: this limitation does not apply to special test assemblies.

RAI 14-10: Core and Fuel Design Dependency of ATWS Instability

Provide the bases and technical justifications that demonstrate [[ ]] response to an ATWS instability event will be bounding in comparison to the response for cores loaded with non-GE fuel, new GE fuel, or mixed cores. Alternatively, provide the licensing limitation that would be necessary for operation along the MELLLA+ boundary, unless specific ATWS instability analyses are provided for cores loaded with non-GE fuel or new GE fuel. Explain what analyses would be required if a plant licensed for operation along the MELLLA+ rodline, was loaded with non-GE fuel (e.g., SVEA 96 or ATRIUM 10) or new Global Nuclear Fuel (GNF) fuel.

See Reference 31 RAI II-1.8 for RAI response.

Evaluation of RAI 14-10:
The NRC staff agrees with GEH's evaluation that the fuel response to ATWS instability is dependant on the assumed fuel design, and that, for plants planning to implement MELLLA+ with a different vendor's fuel or future fuel design beyond GE14, additional justification is required to assure that the core coolable geometry criterion is met with the non-GE fuel design or future GE fuel design.

GEH has committed to perform a confirmatory ATWS instability analysis to justify a different vendor's fuel design or future fuel design beyond GE14. This analysis will be performed with TRACG (or equivalent analytical model) and will simulate the limiting TTWB event resulting in regional oscillation mode. Evaluation of RAI 14-9 provides additional discussion and an associated limitation.

RAI 14-11: Modeling Assumptions - Debris Filters

Were the fuel debris filters modeled in the ATWS analyses? If the fuel debris filters were not included in the analyses supporting MELLLA+ ATWS, explain the reason why the debris filters and the corresponding pressure drops were not included in the analyses. Justify why the results are acceptable. Alternatively, please provide the results of sensitivity analyses that demonstrate the impact of the debris filters on the plant's response to an ATWS. Similar effects should be described for transient analyses.

See Reference 31 RAI II-1.9 for RAI response.

Evaluation of RAI 14-11:

Additional pressure drop caused by the fuel debris filters was not included in the calculation. The NRC staff agrees that lowering the assumed core inlet friction enhances unstable power oscillations and, thus, is a conservative assumption with respect to stability. However, debris filters may reduce the natural circulation state flow after RPT, therefore making the reactor more unstable. Thus, their total effect must be considered.

Limitation:

The plant-specific ATWS calculations must account for all plant- and fuel-design-specific features, such as the debris filters.

NRC RAI 15: INCREASED PROBABILITY OF STARTUP INSTABILITIES

The WNP-2 (Columbia) instability event was caused primarily by an extremely skewed radial power distribution, which was achieved by withdrawing most of the hot-channel control rods early during the startup process. Following the instability event, GEH recommended that hot channel control rods not be withdrawn fully until after the pump up-shift maneuver, when the reactor is more susceptible to startup instabilities. In consideration that a MELLLA+ design core will have significantly more hot channels, two issues need to be addressed:

1. Are the radial power distributions likely to be more skewed during startup (as in the Columbia event) because there are so many hot channels that the operator will have to withdraw the control rods?
2. Will guidance be provided to utilities and operators that startup control rod patterns that have worked in the past may result in instabilities during normal control rod maneuvers?

See Reference 31 RAI II-1.10 for RAI response.

Evaluation of RAI 15:
GEH agrees that the EPU core designs result in a greater number of bundles near the maximum power and adds that the MELLLA+ core design does not, strictly speaking, need higher power peaking than MELLLA core designs. Therefore, operators of both EPU and MELLLA+ plants must take proper care not to increase the probability of startup instabilities

NRC RAI 16: SAFETY SYSTEMS ACTUATION LIMITS

RAI 16-1: NPSH of Safety Systems that Depend on Suppression Pool Water

What are the net positive suction head (NPSH) limits for safety systems that depend on suppression pool water (e.g. RHR, high pressure cooling injection (HPCI), etc)?

See Reference 31 RAI III-1 for RAI response.

Evaluation of RAI 16-1:

NPSH Design for Accidents

For accidents, GEH states that the NPSH limits for safety systems that take suction from the suppression pool during an accident are different for plants with different vintage and their licensing commitments. The plants can be grouped into two categories: Pre-Regulatory Guide 1.1 design (BWR/3 and early BWR/4 plants) and Regulatory Guide 1.1 design (late BWR/4 and BWR/5,6 plants).

Pre-Regulatory Guide 1.1 Design

[[

]]

Regulatory Guide 1.1 Design

For plants in the category, the NPSH design does not depend on containment overpressure credit. It assumes 0 psig containment pressure and the highest suppression pool temperature. For example, [[

]]

NPSH Limit, During Non-Accident Events

For accidents, NRC accepts the NPSH designs with limited or no containment overpressure credits. In the RAI response, GEH states that for non-accident events such as ATWS, NRC does not limit the credit for containment overpressure. Staff does not concur with this statement in that the NRC staff has, thus far, taken explicitly different position with regard to NPSH credit for non-accident events such as ATWS.

For non-accident events such as the ATWS, the NPSH limit depends on the availability of containment overpressure. The NRC staff finds that the containment atmosphere heats up and pressurization rates are different from LOCA and ATWS. The ATWS containment pressurization will occur at a slower rate. Therefore, when containment over-pressure credit is taken for ECCS equipment, the containment calculations must use modeling assumptions that include the slow containment pressurization rate. The LOCA containment conservative assumptions are likely not conservative for ATWS containment over-pressure credits. For example, The STEMP calculations discussed in RAI 16-3 assumes that the peak drywell pressure is equal to the peak wetwell (suppression pool) pressure. This assumption may not be conservative if the NPSH overpressure credit containment analysis is based on similar assumption.

NPSH Limit for HPCI pumps

The NPSH specification for the HPCI pumps requires a suppression pool temperature lower than 140°F, which is based on the HPCI pump lube oil system temperature limit. Since the HPCI operation temperature limit is 140°F, then it is also the NPSH limit for the HPCI pumps. This limit applies to both accident and non-accident events for the high pressure ECCS HPCI system. Other lower-volume equipment can operate with temperatures as high as 212°F.

These numbers are plant-dependent and may depend on containment over-pressurization credits. For example, with 5 psig containment credit, Plant D can operate the HPCI pumps with inlet water up to 170°F for short periods of time. For Plant D, the ODYN suppression pool temperature is 198 F, which occurs at 1770 seconds. The hot shutdown is reported to occur at 1408 seconds. Therefore, NPSH evaluation for all safety systems relied upon to provide makeup water for the event duration is necessary.

Limitation:

Plant-specific applications must review the safety system specifications to ensure that all of the assumptions used for the ATWS SE indeed apply to their plant-specific conditions. The NRC staff review will give special attention to crucial safety systems like HPCI, and physical limitations like NPSH and maximum vessel pressure that RCIC and HPCI can inject. The plant-specific application will include a discussion on the licensing bases of the plant in terms of NPSH and system performance. It will also include NPSH and system performance evaluation for the duration of the event.

RAI 16-2: HPCI Maximum Back-Pressure

The pressure during ATWS events oscillates as high as 1200 psi for long periods (>20 minutes). Is HPCI capable of injecting sufficient volume with such high backpressure? Are any other safety systems affected by a 1200 psi backpressure?

See Reference 31 RAI II-2 for RAI response.

Evaluation of RAI 16-2
HPCI and RCIC are constant flow systems up to the capability of the turbine controls. The maximum operating pressure that is quoted in some documents refer to the maximum pressure below which HPCI and RCIC can deliver full flow. At higher pressures, HPCI and RCIC perform as constant speed centrifugal pumps; therefore, as pressure increases above design, pump flow rates may be expected to decrease accordingly, but the total injection flow rate would still be substantial. Nevertheless, this must be evaluated on plant-specific bases.

RAI 16-3: Containment Pressure

The STEMP results show containment pressurizations as high as 12 psig. Do such high containment pressures affect the actuation of any safety grade systems in the containment such as air-actuated valves?

See Reference 31 RAI II-3 for RAI response.

Evaluation of RAI 16-3
Section 10.3 of the MELLMLTR covers the assessment of environment on the qualification and function of safety systems. The plant-specific application will include confirmation that existing environmental envelopes for safety grade equipment remain valid with EPU and MELLLA+ conditions. The limiting pressures in the containment occur during DBALOCA, which bound the pressures resulting from the ATWS event. GEH provided generic evaluation comparing the DBA-LOCA and ATWS peak drywell and wetwell pressures. Table A-15 provides comparisons of the peak containment and suppression pool pressures.

Table A-15 DBA-LOCA and ATWS Containment and Suppression Pool Peak Pressure Comparisons

	Brunswick MELLLA+/EPU Mark I Containment	Representative Mark II Containment (PU)	Representative Mark III Containment (MELLLA+/EPU)
DBA-LOCA Peak Drywell Pressure (psig)	46.4	39.9	23.2
ATWS Peak Drywell Pressure* (psig)	12.7*	13.7*	7.2*
DBA-LOCA Peak Wetwell Pressure (psig)	31.1	27.9	7.0 **
ATWS Peak Wetwell (psig)	12.7*	13.7*	7.2*

*Note The ATWS analysis assumes that the peak drywell pressure is equal to the peak wetwell pressure calculated with the STEMP code.
** This is the peak pressure which occurs in the containment airspace above the HCU floor.

Limitation:

Plant-specific applications must ensure that an increase in containment pressure resulting from ATWS events with EPU/MELLLA+ operation does not affect adversely the operation of safety-grade equipment.

NRC RAI 17: PLANT E -SPECIFIC REQUEST FOR ADDITIONAL INFORMATION

Plant E is also included in EPU/MELLLA+ reference plants for the evaluating the ATWS response. This section of the RAI addresses specific assumptions used for these analyses.

RAI 17-1: Plant E Suppression Pool Limits

Justify the use of the 185°F ATWS suppression pool temperature limit for the EPU/MELLLA+ ATWS analysis. Specifically, justify why the suppression pool temperature limit is higher than the temperature limit required for depressurization.

See Reference 31 RAI V-1 for RAI response.

Evaluation of RAI 17-1:
GEH chose not to address this question in a generic basis, citing that it is best discussed on a plant-specific basis.

Limitation:

The plant-specific applications must justify the use of plant-specific suppression pool temperature limits for the ODYN and TRACG calculations that are higher than the HCTL limit for emergency depressurization.

RAI 17-2: ATWS Transients Following the EOPs

The peak suppression pool temperature for EPU/MELLLA+ reported in NEDC-33057P is 171° F. While this number is below the reported 185°F limit, the reactor is still at full pressure. Thus, the reported 171° F is not the peak temperature, but the initial condition prior to depressurization. It would appear that following a depressurization (which is required by the EOP at this temperature), the suppression pool temperature would be greater than 185° F. Please provide the actual peak suppression pool temperature when the ATWS transient is followed to completion according to the EOPs.

See Reference 31 RAI V-2 for RAI response.

Evaluation of RAI 17-2:
Even though GEH chose not to answer this RAI, the ATWS transient following the EOPs was provided in the generic TRACG analysis responses. Note: these RAIs apply only to plants with standpipe boron injection. For plants that spray boron in the upper plenum, boron stagnation is not an issue, and the reactor achieves shutdown through borated water spray promptly. However, for BWR5 and 6 (HPCS plants) such as Plant E, the containment limits are significantly smaller and may be breached in the short times that it requires to achieve hot shutdown. Therefore, a plant specific calculation is also required for these plants.

RAI 17-3: EOPs

The effect of EPU/MELLLA+ on EPG/SAGs. Provide a critical review of the EPGs/SAGs to determine which variable definitions and calculations are affected by EPU/MELLLA+. The following sections provide some examples of areas that need further evaluation and update for determining limiting values. Other variables not mentioned here may be affected.

1. Maximum Pressure for Heat Capacity Temperature Limit Plot (Section 17.5). Section 17.5 defines the procedure for calculation of the HCTL. In the example plots (Figs. B-17-5 and B-17-6) a maximum pressure of 1100 psig is used. However, TRACG calculations show that the pressure during an MSIV ATWS is consistently above 1100 psig. Should the EPG/SAGs be modified for EPU/MELLLA+ operation to require calculation of the HCTL at the expected higher pressures?
2. Hot Shutdown Boron Weight (Section 17.6). The first assumption is that the reactor is operating on the maximum extended operating domain. Clearly, this assumption should be changed to the corner of the MELLLA+ domain. Assumption #6 specifies an operating pressure of 1100 psia. However, TRACG calculations show that during ATWS under EPU/MELLLA+ conditions the expected pressures are significantly higher than 1100 psia.
3. Minimum Number of SRVs Required for Decay Heat Removal (Section 17.21). With EPU/MELLLA+, the expected decay heat levels should be higher. Will the minimum number of SRVs change? If the minimum of SRVs does change, will this affect any other variables?
4. Minimum Number of SRVs Required for Emergency Depressurization (Section 17.22). With EPU/MELLLA+, the expected ATWS power levels should be higher. Will the minimum number of SRVs change? If the minimum number of SRVs does change, will this affect any other variables?
5. Minimum Steam Cooling Pressure (Section 17.23). With EPU/MELLLA+, the expected ATWS power levels should be higher. Will the minimum steam cooling pressure change? Will this pressure change affect any other variables?

6. Minimum Steam Cooling RPV Water Level (Section 17.24). With EPU/MELLLA+, the expected ATWS power levels should be higher. Will the minimum steam cooling RPV water level change? If the level does change, will this affect any other variables?
7. Minimum Zero-Injection RPV Water Level (Section 17.25). With EPU/MELLLA+, the expected ATWS power levels should be higher. Will the minimum zero-injection RPV water level change? If the water level changes, will this affect any other variables?

See Reference 31 RAI V-4 for RAI response.

Evaluation of 17-3:
GEH chose not to address this question on generic bases, citing that it is best discussed on a plant specific basis.

APPENDIX B

NRC STAFF EVALUATION OF FUEL DEPENDENT ANALYSES RAI RESPONSES

TO

SAFETY EVALUATION BY

THE OFFICE OF NUCLEAR REACTOR REGULATION

LICENSING TOPICAL REPORT NEDC-33006P

"GENERAL ELECTRIC BOILING WATER REACTOR MAXIMUM EXTENDED LOAD LINE LIMIT ANALYSIS PLUS"

GENERAL ELECTRIC HITACHI NUCLEAR ENERGY AMERICA, LLC

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APPENDIX B NRC STAFF EVALUATION OF FUEL DEPENDENT ANALYSES RAI RESPONSES

This appendix provides the NRC staff's evaluation of responses to requests for additional information (RAIs). This appendix only provides the RAI question and evaluation, not the RAI response. The RAI responses can be found in References 17, 18, 19, 20, 21, 22, 23, 28, 29, 30, 32, 39, and 44.

NRC RAI 1: TIME VARYING AXIAL POWER SHAPES (TVAPS)

a. [[

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b. (Based on the audit). Provide a background discussion on why the fuel channels experience axial power shape changes during pressurization transients.

c. What are the principle factors that control the severity of the change in the critical power ratio (ΔCPR) response to TVAPS. Does the severity of the critical power ratio (CPR) change with TVAPS increase for the extended power uprate (EPU)/maximum extended load line limit analysis plus (MELLLA+) operating condition? Explain the impact of the EPU/MELLLA+ condition on the factors that control the severity of the CPR change due to TVAPS effect. Would the effect of TVAPS on the ΔCPR be more severe for 55% core flow (CF), 80% CF, 100% CF along the MELLLA+ upper boundary or the EPU/increased CF (ICF) as an initial condition. Does the severity of the TVAPS effect on the CPR differ for different pressurization transient?

d. Amendment 27 to GESTAR II (submitted for NRC staff review) states that "NRC-agreed upon methodology for evaluating GE11 and later fuel uses TVAPS, thereby changing the need for assuring this check. See GENE-666-03-0393 and NRC staff agreement at meeting on April 14, 1993." Explain this statement and state if the NRC reviewed and approved the method used to check or account for the effect of TVAPS on the CPR change during pressurization transients.

e. If the method used to evaluate the effect of TVAPS during a pressurization transient was not reviewed by the NRC staff in the supplement to Amendment 27, provide sufficient information, including sensitivity results so that the NRC staff can review the method and the effects of TVAPS on the transient response for plants operating with the EPU/MELLLA+ core design.

See References 28 and 30 for RAI responses.

Evaluation of RAI 1:
The GEH licensing methodology in GESTAR II does not provide sufficient evaluations that address both the effects of TVAP and the adequacy of how it is accounted for. In addition, it did not appear that the NRC staff specifically reviewed or approved the methodology used to address the TVAP effects. Therefore, the intent of this RAI was to understand the TVAP effects and how the operation at high bundle power at lower flow conditions affects the severity of the TVAP.

GEH states that although the NRC did not formally review and approve the method to check or account for the effects of TVAP on CPR change during pressurization response, NRC was informed. Subsequently, NRC staff covered the TVAPS effects in the GE11 audit on March 1992. GEH added that the inclusion of the TVAP effects in the analysis represented

conservative change, which is allowed under the Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.59 process. The NRC staff notes that GEH is not a licensee, and thus the 10 CFR 50.59 process may not apply. However, GEH has informed the NRC in letter dated November 5, 1991 (MFN 150-91, "Pressurization Transient Analysis Procedure for GE11.") Therefore, the NRC staff finds although it is not explicitly approved, NRC had the opportunity to review the TVAP effects and the NRC staff finds the justification provided acceptable.

The RAI response described the TVAP phenomena and presents the changes in the maximum bundle thermal-hydraulic conditions due to the TVAP for pre-EPU and EPU/MELLLA+ as follows:

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Figure B-1 shows the changes in the axial power shape as the control rod inserts for a maximum powered bundle. As can be seen from Figure B-1, The severity of the axial power peaking and the axial power shape change, both of which affect the MCPR response. Higher bottom-peak or double-hump power shapes of partially controlled cell can make the axial power shift more pronounced. However, the TVAP effects will be included in the EPU/MELLLA+ pressurization transients. Thus, the impact of the axial power shapes resulting from operation at reduced flow and spectral shift operation will be accounted in the analysis methodology.

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Figure B-1 BWR/4 Hot channel Transient Varying Axial Power Shape

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Figure B-2 Changes in Maximum powered bundle Mass flux for TTNBP event

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Figure B-3 Maximum Powered Bundle CPR Response Changes

NRC RAI 2: TVAPS EFFECT FOR PLANT D

For the Plant D EPU/MELLLA+ analyses, explain what method would be used to calculate TVAPS. According to the proposed Amendment 27 changes to Section 4.3.1.2.1 of GESTAR, the TVAPS for GE11 fuel and later products is calculated using ODYN. The NRC staff has been informed that Plant D is using TRACG to perform the EPU/MELLLA+ reload analysis. As such, how does ODYN interface with TRACG? Based on the Plant D EPU/MELLLA+ core, provide a description of how the TVAP effect on the CPR was accounted for and calculated. Provide plots of the results.

See References 28 and 29 for RAI responses.

Evaluation of RAI 2:
The RAI response provided time histories of changes in the hot channels and core-wide parameters. The response did not contain comparisons for changes in the parameters for operation at different statepoints (EPU, MELLLA, and MELLLA+) or different axial power distribution, in order to gage changes in the severity of TVAP. However, as stated in RAI 1, the effects of TVAP will be accounted for in the analysis methodology.

NRC RAI 3: [[

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See References 30 and 32 for RAI responses.

Evaluation of RAI 3:
Global Nuclear Fuels (GNF) [[

]] GNF response states that water rod modeling will be included in future TRACG analyses. NRC staff finds this appropriate and acceptable.

The RAI also inquired about other codes, including ISCOR, PANACEA, ODYN, and TASC. GNF states the following with respect to water rod modeling in these codes: [[

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Therefore, PANACEA and ODYN codes account for the appropriate water rod and bypass flows and NRC staff finds this acceptable. TRACG is the preferred code to be used for peak cladding temperature (PCT) calculations upon NRC approval of the code.

Limitation: (Anticipated Operational Occurrence (AOO) RAI 3)

For EPU/MELLLA+ plant-specific applications, that use TRACG or any code that has the capability to model in-channel water rod flow, the supporting analysis will use the actual flow configuration.

NRC RAI 4: EFFECTS OF BYPASS VOIDING

See References 28 and 30 for RAI and responses.

Evaluation of RAI 4:
The NRC staff review of bypass voiding is covered in NEDC-33173P, which includes the applicable limitations.

NRC RAI 5: BYPASS VOIDING FOR PLANTS D AND E

See References 28 and 30 for RAI and responses.

Evaluation of RAI 5:
The NRC staff review of bypass voiding is covered in NEDC-33173P, which includes the applicable limitations. Table B-1 below shows the calculated bypass voiding at the LPRM D level. The bypass voiding will be calculated on plant-specific bases.

Table B-1 D-level Bypass Void Fraction

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However, the licensee would calculate the potential for bypass voiding during steady state for plants licensed with EPU/MELLLA+ operation. These analyses will be performed using conservative models and code systems that do not underestimate the potential for bypass voiding. In addition, the core and/or 4 bundles with bypass configuration used to simulate the physical phenomena would be based on conservative operating conditions that bound the expected core thermal-hydraulic conditions. Some of the key parameters or assumptions necessary in order to ensure conservative assumptions, include the number of high powered bundles in the 4 bundle with bypass configuration, the most limiting statepoint, and the most limiting control rod pattern that would lead conservative power distribution.

NRC RAI 6: VOID FRACTIONS GREATER THAN 90 PERCENT

The Brown Ferry steady state TRACG analysis shows that the hot channel exit void fraction is greater than 90 percent. This could potentially affect the validity of the exit conditions assumed in the computational models used to perform the safety analyses. The audit documents indicate that GEH had evaluated the effect of the high exit void fraction on the analytical models, techniques and methods. However, the evaluations and the bases of the conclusions were not discussed in the MELLLA+ LTR or submitted for NRC review as an amendment to GESTAR II.

The following RAIs address the effect of the high exit void fraction and quality on the EPU/MELLLA operation.

a) Provide an evaluation of the analytical methods that are affected by the hot channel high exit void fraction (>90 percent) and channel exit quality. Discuss the impact the active channel exit void fraction would have on:

   i. the steady-state nuclear methods (e.g., PANAC/ISCOR),
   ii. the transient analyses methods (e.g., ODYN/TASC/ODSYS),
   iii. the GEXL correlation, and
   iv. the plant instrumentation and monitoring.

b) Evaluate whether the higher channel void fraction would affect any benchmarking or separate effects testing performed to assess specific thermal-hydraulic and/or neutronic phenomena.

c) Include in your evaluation, the effect of the high void fractions on the accuracy and assessment of models used in all licensing codes that interface with and/or are used to simulate the response of BWRs, during steady state, transient, and accident conditions.

d) Submit an amendment to the appropriate NRC-approved codes (e.g., TRACG for AOO, ODYN/ISCOR/TASC, SAFER/GESTR/TASC, ODSYS) that updates and evaluates the impact of the EPU/MELLLA+ operating conditions such as the high exit void fraction on the computational modeling techniques and the applicability range.

e) Submit a supplement to the MELLLA+ LTR that addresses the impact of the EPU/MELLLA+ core operating conditions, including high exit void fraction, on the applicability of the currently approved licensing methods.

See References 19, 21, 22, 28, and 30 for RAI responses.

Evaluation of RAI 6:

In response to this RAI, GEH submitted Enclosure 3, "Applicability of NRC Approved Methodologies to MELLLA+," (MFN-04-026). Enclosure 3 referred to as the Methods LTR technical evaluation of key technical models used within the NRC licensed methodologies and justified the applicability of the extension of the GEH methodologies to MELLLA+ core conditions. The Enclosure justified extension of the steady-state nuclear methods to high void conditions. The accuracy of the neutronic methods affects all methods employed by GEH. Enclosure also addressed several other methods topics that may be extended outside the applicability ranges.

The NRC staff reviewed the Methods LTR and issued RAIs. GEH provided partial RAI response. The NRC staff determined that the issues considered in the Enclosure were not limited to MELLLA+ application, but were relevant to EPU core conditions. The NRC staff also determined that the benchmarking of the extension of the neutronic methods to high void conditions, provided in Enclosure 3 were not sufficient. In order to establish the bundle and pin power uncertainties, validations against measurement data was necessary (e.g., gamma scans). Subsequently, the methods topics were evaluated and resolved as an interim measure for a plant-specific EPU application. GEH submitted NEDC-33173P, which paralleled the interim

approach implemented in the plant-specific EPU application. The NRC staff reviewed and approved NEDC-33173P and the associated limitation are provided in the associated NRC staff SER.

GEH had committed to performing the gamma scans to benchmark the neutronic methods. The topics and the associated RAI response to Enclosure 3, if not resolved under NEDC-33173P will be incorporated in the review of the gamma scan data

NRC RAI 7: PLANTS D AND E - EFFECT OF VOID FRACTIONS GREATER THAN 90 PERCENT

a. Explain how the core averaged void fraction reported in the heat balance table is computed. For example, the Plant D MELLLA+ application reports core averaged void fractions in the range of 0.51 to 0.54 for different statepoints.

b. For the EPU/MELLLA+ core design, what is the hot channel exit void fraction for the steady state operation at the EPU 120 percent power/99 percent CF, EPU/MELLLA+ 120 percent power/85 percent CF and the EPU/MELLLA+ 77.6 percent power/55 percent CF statepoints? Use bounding conditions.

See References 28 and 30 for RAI responses.

Evaluation of RAI 7:
The RAI response provided that the active coolant average void fraction, excluding the unheated and bypass regions.

$$<VF> = \frac{\sum_{i=1}^{\#\,channel\,types} n_i \frac{\sum_{k=1}^{\#\,nodes} VF_k \, FlowArea_k}{24 < FlowArea >}}{Total\ \#\ of\ Bundles}$$, where i is the ISCOR channel types and k is the axial nodes.

The core averaged void fraction generally reported in EPU applications as means to compare with thermal-hydraulic conditions for the current licensed against the uprated condition. Similarly, the Plant D EPU/MELLLA+ application reported the core averaged void fraction, which did not give adequate indication of the void distribution for the high powered bundles. Therefore, the exit void conditions for the high powered bundle would be a better indicator of the EPU/MELLLA+ thermal-hydraulic conditions. Table B-2 provides the exit voids for Plant D at MELLLA+ statepoints.

Table B-2 Plant D Exit Void Fraction

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The NRC staff finds the response acceptable. The applicable limitation is provided below.

Limitation: (AOO RAI 7)
The EPU/MELLLA+ application would provide the exit void fraction of the high-powered bundles in the comparison between the EPU/MELLLA+ and the pre-MELLLA+ conditions.

NRC RAI 8: ICF

Are the shutdown margins, standby liquid control system (SLCS) shutdown capability and mislocated fuel bundle analyses performed at the rated conditions (100 percent EPU power/100 percent CF). If so, justify why these calculations are not performed for the non-rated conditions such as the ICF condition. Provide supporting sensitivity analysis results for your conclusions or update the GESTAR II licensing methodology, stating that these calculations would be performed at the ICF statepoint.

See References 28 and 30 for RAI responses.

Evaluation of RAI 8:
The intent of the RAI was to determine how the calculation of peak reactivity, during the cycle for the shutdown margin (SDM) and SLCS methods, account for operation at different statepoints. For example, in identifying the cycle statepoint where the peak reactivity occurs, are the plant-specific operating history considered or depletion at rated conditions assumed. For MELLLA+, plants will be operating with spectral shift, operating at the reduced CF statepoints and increasing flow as the core depletes. In addition, are the mislocated fuel bundle analysis performed at rated conditions and why would this be considered to be the bounding statepoint? The RAI response clarified that the SDM, SLCS capability to achieve cold shutdown condition and the mislocated fuel bundle analysis are performed [[

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Therefore, these analyses would be performed, accounting for the core configuration at the low flow EPU/MELLLA+ condition.

NRC RAI 9: ICPR CALCULATED FROM OFF-RATED CONDITIONS

The hot channel void fraction increases with decreasing flow along the MELLLA+ upper boundary. Therefore, the void fraction at the 55 percent CF and the 80 percent CF statepoints are higher than the void fraction at 99 percent CF. Consequently, it is feasible that the initial conditions of the hot channels could be higher at the minimum CF statepoints or at the off-rated conditions.

a. Justify why the steady-state ICPR is assumed in determining the off-rated AOO response, instead of the ICPR calculated from off-rated conditions.

b. For the most bounding conditions, compare the steady-state ICPR calculated based on the actual conditions at the statepoints (rated, 80 percent CF, and 55 percent CF,or off-rated lower power and flow conditions).

See References 28, 30, and 32 for RAI responses.

Evaluation of RAI 9:

The objective of the RAI was to determine if the initial conditions assumed in the AOO analyses for the high power/low flow conditions accurately reflect the limiting thermal-hydraulic conditions at these statepoints, instead of assuming that the rated power conditions are most limiting in terms of power distribution and the associated core thermal-hydraulic conditions.

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The tables below show the ICPR associated with the results in Table 9-2 of the M+ LTR for the rated power (at minimum and rated flow) and the off-rated power (55% CF) conditions. The tables do show that higher operating limit MCPR (OLMCPR) is assumed for the off-rated power/flow conditions (55% CF).

Table B-3 AOO Off-rated ICPR

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Table B-4 AOO Rated ICPR

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NRC RAI 10: ISCOR/ODYN/TASC APPLICATION

The transient CPR and the peak cladding temperature (PCT) calculations are performed using the ODYN/ISCOR/TASC combination. The NRC staff understands that ISCOR calculates the initial steady-state thermal-hydraulic core calculations. ODYN (1-D code) provides the reactor power, heat flux, CF conditions, and the axial power shapes of the hot bundle during the transient. [[

]] The ISCOR/TASC combination is also used to calculate the PCT for emergency core cooling system (ECCS)-loss-of-cooling accident (LOCA) and Appendix R calculations. In addition, ISCOR/TGBLA/PANAC code combinations are also used in core and fuel performance calculations.

a. ISCOR is widely used in many of the safety analyses, but the code was never reviewed by the NRC. The use of a non-NRC-approved code in a combined code system applications is problematic. Therefore, submit the ISCOR code for NRC review.

b. Although ISCOR is not an NRC-approved code, our audit review did not reveal specific shortcomings. [[

]] Therefore, include in the ISCOR submittal a description and evaluation of the ISCOR/ODYN or

ISCOR/TGBLA/PANAC code combination discussed above. Provide sufficient information in the submittal, including sensitivity analyses, to allow the NRC staff to assess the adequacy of these combined applications.

c. During the MELLLA+ audit , the NRC staff discovered that GEH had internally evaluated a potential non-conservatism that may result from the use of the flow-driven ISOR/ODYN/TASC combination to calculate the transient ΔCPR. [[

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See References 30 and 32 for RAI responses.

Evaluation of RAI 10:
The RAI responses covered GEH=s bases for concluding that the NRC-approved ODYN flow driven method is acceptable in comparison to the more conservative ODYN pressure driven method. [[

]] Figure B-4 below compares the results from the two codes [[

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Figure B-4 ODYN/TASC versus TRACG Comparisons

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The NRC staff concurs with GEH that for MELLLA+ most licensees may elect to transition to TRACG for AOOs, considering that only TRACG can be used for the ATWS analysis. TRACG is has more detailed modeling capabilities of the reactor conditions, thus representing better modeling of the physical phenomena of boiling water reactors (BWRs). GEH/GNF had also provided several references in which ISCOR/ODYN/TASC code combination had been covered in the LTRs under NRC-approval. The NRC staff concludes that the use of TRACG as oppose to ODYN is acceptable approach.

Although ISCOR was not explicitly approved, the ISCOR/ODYN/TASC code combination approach was use in the GEH historical codes reviewed and approved in the past. The references that describe the use of ISCOR in the GNF methodology follow.

1. General Electric Standard Application for Reactor Fuel, GESTAR II, NEDE-24011-P-A-14, June 2000.

2. General Electric Standard Application for Reactor Fuel (Supplement for United States), NEDE-24011-P-A-14-US, June 2000.

3. Steady State Nuclear Methods, NEDE-30130-P-A, April 1985.

4. TASC-03A Computer Program for Transient Analysis of a Single Channel, NEDC-32084PA, July, 2002.

5. Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors. NEDO-24154-A, Volume I, August 1986.

6. Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors. NEDO-24154-A, Volume II, August 1986

7. Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors. NEDE-24154-P-A, Volume III, August 1988

Based on the potential for GEH to transition to the TRACG for EPU/MELLLA+ application and the reasons cited above, the NRC staff agrees that NRC review and approval of ISCOR is not necessary for EPU/MELLLA+ applications. The NRC staff finds the response acceptable.

NRC RAI 11: PLUTONIUM BUILDUP

It is expected that a EPU/MELLLA+ core would produce more Pu(239). What are the consequences of this increase from a neutronic and thermal-hydraulic standpoint during steady state, transient, and accident conditions?

See References 28 and 30 for RAI responses.

<u>Evaluation of RAI 11:</u>
In the RAI Response, GNF stated the following.

> The core simulator will properly capture any resulting increase of plutonium from high void operation. Additionally, the cycle specific transient analyses consider variation on

the burn strategy and Pu production by varying the degree at which the bottom of the core is burned early in the cycle. Therefore, any changes in isotopic inventory because of MELLLA+ operation will be explicitly modeled for the purposes of determining cycle specific analyses including selection of rod patterns, safety evaluations (SDM), transient evaluations, as well as others.

The NRC staff expected the RAI response would provide some explanation of changes in the Pu production for EPU/MELLLA+ core would be different from the pre-uprate conditions.

However, the impact of spectral shift operation at EPU/MELLLA+ conditions were covered in the review of LTR NEDC-33173P. In the RAI responses associated with Enclosure 3 (MNF 04-026), the NRC staff had asked GEH to provide the isotopics generated for operation at different void conditions expected bundles to deplete under at different elevations, The NRC staff also generated lattice physics data that demonstrate the changes in the isotopics with voids. Therefore, although the RAI response is inadequate, the related issues were reviewed and resolved under NEDC-33173P.

NRC RAI 12: SPECTRUM HARDENING

How does the harder spectrum from the increased Pu affect surrounding core components such as the shroud, vessel, and steam dryer?

See References 28 and 30 for RAI responses.

Evaluation of RAI 12:
GNF provided a discussion of the affect of the increased Pu on the surrounding core components. The extent of the impact would be covered on plant-specific evaluation. The NRC staff finds the response acceptable.

NRC RAI 13: THERMAL MARGINS UNDER EPU/MELLLA+ OPERATION

How do the thermal margins change as a function of flow and transients for a EPU/MELLLA+ cores?

See References 28 and 30 for RAI responses.

Evaluation of RAI 13:
GEH provided TRACG ΔCPR/ICPR for Plant D initiated from different power/flow conditions. For the limiting pressurization transients, the low power/high flow cases result in higher thermal margin changes. However, it is not clear if consistent ICPR were applied in order to make comparisons of the actual ΔCPR change that will yield the most limiting OLMCPR value.

Table B-5 AOO ΔCPR/ICPR Results

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NRC RAI 14: ROD WITHDRAWAL ERROR (RWE)

Demonstrate that the RWE for the EPU/MELLLA+ domain is less limiting than the non-MELLLA+ domain throughout the cycle.

See References 28, 29, and 30 for RAI responses.

Evaluation of RAI 14:

Table B-6 presents [[

]] Table B-7 provides similar confirmation RWE analysis for Plant D performed at the EPU/MELLLA+ minimum CF statepoint and at rated EPU conditions.

The RAI response states that RWE results show are no sensitivity to CF. Although the data does not show trend with flow, it does show that [[

]] RWE analysis will be performed to confirm the rod block monitor (RBM) setpoints. Section 9.1.1.2 of this SE provides additional discussion and an associated limitation.

Table B-6 Generic ARTS RWE

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Table B-7 EPU/MELLLA+ RWE

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NRC RAI 15: EFFECT OF AXIAL POWER SHAPE ON TRANSIENT RESPONSE

If the axial power profile is expected to be more pronounced (i.e., more limiting) for EPU/MELLLA+ core, demonstrate and provide a quantitative and qualitative technical justification of the effects of these more pronounced profiles on the normal and transient behavior of the core.

See References 28 and 30 for RAI responses.

Evaluation of RAI 15:
GNF states that the Plant D EPU/MELLLA+ core power distribution does not indicate any change in the transient response due to axial power profile. Since the plant-specific application will provide thermal limits assessment, the response is acceptable.

NRC RAI 16: RELOAD ANALYSES

Since the startup and intermediate rod patterns are developed by the licensees and subject to change during plant maneuvers, explain how you ensure that the core and fuel assessment analyses performed during the reload are still applicable. For example, if the SLMCPR is performed at different burn up conditions during the cycle, how do you ensure that the plant=s operating history does not invalidate the reload assumptions? How are the corrections or adjustments made to the plant=s core and fuel performance analyses to ensure the parameters and conditions assumed during the reload analyses remain applicable during the operation. The NRC staff=s concern stems from the additional challenges that the EPU/MELLLA+ operation poses in terms of core and fuel performance.

See References 29 and 30 for RAI responses.

Evaluation of RAI 16:
The RAI response described how it is ensured that any deviation from the planned cycle operation does not inviolate the conditions assumed in the reload analysis. The NRC staff finds the described process acceptable. However, the RAI response also states that the design rod patterns represent a relatively detailed simulation of core operation at rated power using an operational philosophy that incorporates any utility instructions (regarding how they intend to operate). For EPU/MELLLA+ conditions, the NRC staff is concerned that the operation at the 120% power/85% CF or the 55% CF statepoints, the rod patterns assumed in these analyses

may not be part of the process used to ensure that the plant is operated within the limiting rod patterns assumed at the minimum CF or off-rated statepoint. This is of concern for EPU/MELLLA+ conditions, because the all-rod-out condition near the end of cycle (EOC) may no longer be the limiting condition.

The objective is to ensure that the plant is not operated with power distributions that would be more limiting than assumed in the analyses. The conservatism of the assumed rod patterns for the calculation of SLMCPR at minimum CF statepoint is important. While the rod patterns assumed in the rated conditions were reviewed and accepted in the NRC staff review of NEDC-32601P-A and NEDC-32694P-A, for operation at minimum CF statepoint, the limiting control rod patterns were not reviewed and approved. GEH had committed to submit updated SLMCPR methodology. In the interim, the NRC staff reviews the bounding control rod patterns used on plant-specific bases. The control rod patterns assumed in the transient analyses are addressed in NEDC-33173P review.

NRC RAI 17: THERMAL LIMITS ASSESSMENT

a. SLMCPR. It is possible that the impact on the critical heat flux (CHF) phenomena may be higher at the off-rated or minimum CF statepoints. Is the SLMCPR value provided in the SLMCPR amendment requests and reported in the technical specification (TS) based on the rated conditions? If so, justify why the SLMCPR is not calculated for statepoints other than the rated conditions. Quantitatively demonstrate that the SLMCPR calculated at the minimum 80 percent and 55 percent statepoints would be lower than the SLMCPR calculated at the rated conditions. Use power profiles and core designs that are representative of the EPU/MELLLA+ conditions. Discuss the assumptions made. Include the Plant D EPU/MELLLA+ application in your sensitivity analyses.

b. SLMCPR at EPU/MELLLA+ Upper Boundary. The SLMCPR at the non-rated conditions (EPU power/80 percent CF) could be potentially higher than the SLMCPR at rated conditions, explain how "statepoint-dependent" SLMCPR would be developed and implemented for operation at the EPU/MELLLA+ condition. Use the Plant D EPU/MELLLA+ application to demonstrate the implementation of "statepoint-dependent" SLMCPR.

c. Exposure-Dependent SLMCPR. Discuss the development of the exposure-dependent SLMCPR calculation. State whether this is an NRC-approved method and refer to the applicable GESTAR II amendment request.

See References 29, 30, 32, and 39 for RAI responses.

Evaluation of RAI 17:
In letter dated August 24, 2004, 04-081, "Part 21 Reportable Condition and 60–Day Interim Report Notification: Non-conservative SLMCPR," (Reference 39), GEH states that the SLMCPR at the minimum flow statepoint for the MELLLA operation may be bounding. Four operating cycles were identified as affected. However, GNF also states that the Plant D EPU/MELLLA+ SLMCPR calculation indicates that the minimum CF statepoint and the 55% CF statepoint are bounded by the rated condition. The current GNF methodology is silent on calculating the SLMCPR on statepoint basis.

The Part 21 evaluation stated that the power distribution, resulting from operation at the reduced flow conditions, could yield SLMCPR values that bound the rated SLMCPR value. Subsequently, GEH revised its SLMCPR methodology, including calculation of the SLMCPR at minimum CF in the licensing process. The calculated SLMCPR at the minimum CF statepoint (OLTP/75%F or 105%P/82%F) for several BWRs resulted in a higher SLMCPR value than at the rated conditions. The current GEH SLMCPR applies higher off-rated CF uncertainty for non-rated conditions. In the updated, MFN 07-041(Reference 32), GEH proposes reducing the CF uncertainty applied to the lower CF statepoints, which will result in reduced SLMCPR response.

However, changes in the SLMCPR methodology for reduced flow statepoints including the MIP criterion for operation at the MELLLA+ conditions, the conservatisms of the limiting control rod patterns in relative to the patterns employed at the plants have not been reviewed or approved generically. Currently for reduced CFs, these assumptions are addressed on plant-specific bases. In addition, GEH is evaluating gamma scan data that will benchmark the bundle and pin power distribution uncertainties. These uncertainties factor into the SLMCPR methodology. The NRC staff had requested that GEH submitted updated SLMCPR LTRs for the current and proposed operating strategies. Therefore, any reduced CF uncertainties currently applied to the SLMCPR calculations for operation at the minimum CF statepoints will be reviewed under the revised SLMCPR methodology. As discussed in Section 2.2.1.1 of this SE, the higher off-rated CF uncertainty will be applied to the SLMCPR at the minimum and 55% CF statepoint, until such time the GEH submits the revised SLMCPR methodology.

Therefore the NRC staff concludes that for MELLLA+ core, cycle specific SLMCPR analysis must account for the potentially limiting statepoints, covering lower flow conditions. Section 2.2.1.1 of this SE provides additional discussion and an associated limitation.

NRC RAI 18: GEXL-PLUS CORRELATION

Confirm that the GEXL-PLUS correlation is still valid over the range of power and flow conditions of the EPU/MELLLA+ operations.

See References 28 and 30 for RAI responses.

Evaluation of RAI 18:
Section 1.1.4 of this SE provides additional discussion and an associated limitation. Additionally, this topic is covered in the NRC staff SE of LTR NEDC-33173P.

NRC RAI 19: USING ATWS-RECIRCULATION PUMP TRIP (RPT) FOR AOOS

GEH licensing methodology allows using anticipatory ATWS-RPT in some AOO transients to decrease the power and pressure response. Therefore, the anticipatory RPT is used in some plants to minimize the impact of the pressurization transient on the ΔCPR response. For the EPU/MELLLA+ operation, RPT may subject the plant to instability. Evaluate the runbacks associated with the AOOs and demonstrate that the scram and the RPT timings would not lead to an AOO transient resulting in an instability.

See References 28 and 30 for RAI responses.

Evaluation of RAI 19:
GNF stated that [[

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The NRC staff agrees with GNF that if the scram occurs within 2 seconds, then there is less of concern that the RPT feature would increase the potential for instability event.

NRC RAI 20: MECHANICAL OVERPOWER (MOP) AND THERMAL OVERPOWER (TOP)

Are the fuel-specific mechanical and thermal overpower limits determined based on the generic fuel design or for each plant-specific bundle lattice design? How is it confirmed that the generic MOP and TOP limits for GE14 fuel bounds the plant-specific GE14 lattice designs intended to meet the cycle energy needs at the EPU/MELLLA+ conditions?

See References 28 and 30 for RAI responses.

Evaluation of RAI 20:
The RAI response stated that [[

]] This topic is covered in detail in NEDC-33173P NRC staff evaluation. The NRC staff finds the response acceptable.

NRC RAI 21: PLANT D AOO

The Plant D Units 1 and 2 are the first plants to apply TRACG for performing the reload analyses.

a. Compare the Plant D EPU and the EPU/MELLLA+ core designs and performance.
b. State what the benefit of using TRACG instead of ODYN is for the EPU/MELLLA+ reload analyses.
c. Provide a comparison of the TRACG and ODYN AOO analyses results based on the EPU/MELLLA+ core design.

See References 28 and 30 for RAI responses.

Evaluation of RAI 21:

The RAI response stated that [[

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The RAI response did provide comparisons of TRACG MCPR Operating Limits and ODYN stating that the TRACG OLMCPR is [[ ]] than the corresponding ODYN limits. GEH states that this difference is considered to be a significant thermal margin benefits. Figure B-5 through Figure B-9 compare calculations of key parameters such as neutron flux, CF, vessel pressure and steam flow using TRACG and ODYN. The figures show that ODYN is not significantly more conservative in all instances for the duration of the event. However, the RAI response did not provide the hot bundle conditions, which may have significant differences since ODYN models average bundle conditions. Overall, the two codes are consistent in terms of core wide response parameters except for neutron flux and to a lower degree vessel pressure.

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Figure B-5 Neutron Flux comparison

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Figure B-6 CF Comparisons

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Figure B-7 Steam Flow Comparisons

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Figure B-8 Vessel Dome Pressure CF Comparisons

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Figure B-9 SRV Flow Comparisons

NRC RAI 22: PLANT D AOO DATA REQUEST

See References 17, 23, 28 and 30 for RAI responses and data.

Evaluation of RAI 22:
The requested data was provided.

NRC RAI 23: SEPARATE EFFECTS, MIXED VENDOR CORES AND RELATED NRC STAFF LIMITATIONS

Separate effects: revise Section 1.0, "Introduction," of the MELLLA+ LTR and remove the list of "separate effects" changes. The MELLLA+ LTR lists plant-specific operating condition changes that could be implemented concurrently with the EPU/MELLLA+, but would be evaluated in a separate submittal. All of these lists of changes would affect the safety analyses that demonstrate the impact of EPU/MELLLA+ on the plant=s response during steady-state, transients, accidents, and special events. The plant-specific EPU/MELLLA+ application must demonstrate how the plant would be operated during the implementation of MELLLA+. In addition, the EPU/MELLLA+ reduces the available plant margins. Therefore, the NRC staff cannot make its safety finding based on assumed plant operating conditions that are neither bounding nor conservative relative to the actual plant operating conditions. Revise the MELLLA+ LTR and delete the paragraphs that propose evaluating additional operating condition changes in a separate submittal while the EPU/MELLLA+ application assumes that these changes would not be implemented.

Add the following statements in the MELLLA+ LTR to address NRC staff limitations including: (1) the implementation of additional changes concurrent with EPU/MELLLA+, (2) the applicability of the generic analyses supporting the EPU/MELLLA+ operation, and (3) the approach used to support new fuel designs or mixed vendor cores.

The plant-specific analyses supporting the EPU/MELLLA+ operation will include all planned operating condition changes that would be implemented at the plant. Operating condition changes include but are not limited to increase in the dome pressure, maximum CF, increase in the fuel cycle length, or any changes in the currently licensed operation enhancements. For example, with increase in the dome pressure, the ATWS analysis, the American Society of Mechanical Engineers (ASME) overpressure analyses, the transient analyses, and the ECCS-LOCA analysis must be reanalyzed based on the increased dome pressure. Any changes to the safety system settings or actuation setpoint changes necessary to operate with the increased dome pressure should be included in the evaluations (e.g., safety relief valve setpoints).

For all of the principal topics that are reduced in scope or generically dispositioned in the MELLLA+ LTR, the plant-specific application will provide supporting analyses and evaluations that demonstrate the cumulative effect of EPU/MELLLA+ and any additional changes planned to be implemented at the plant. For example, if the dome pressure would be increased, the ECCS performance needs to be evaluated on a plant-specific basis.

1. Any generic sensitivity analyses provide in the MELLLA+ LTR will be evaluated to ensure that the key input parameters and assumptions used are still applicable and bounding. If the additional operating condition changes affects these generic sensitivity analyses, a bounding generic sensitivity analyses will be provided. For example, with increase in the dome pressure, the TRACG ATWS sensitivity analyses that model the operator actions

(e.g., depressurization if the heat capacity temperature limit is reached) needs to be reanalyzed, using the bounding dome pressure condition.

2. If a new GE fuel or another vendor=s fuel is loaded at the plant, the generic sensitivity analyses supporting the EPU/MELLLA+ condition will be reanalyzed. For example, the ATWS instability analyses supporting the EPU/MELLLA+ condition are based on the GE14 fuel response. New analyses that demonstrate the ATWS stability performance of the new GE fuel or legacy fuel for the EPU/MELLLA+ operation needs to be provided. The new ATWS instability analyses can be provided as supplement to the MLTR or as an Appendix to the plant-specific application.
3. If a new GE fuel or another vendor=s fuel is loaded at the plant, analyses supporting the EPU/MELLLA+ application will be based on core specific configuration or bounding core conditions. In addition, any principle topics that are generically dispositioned or reduced in scope will be demonstrated to be applicable or new analyses based on the transition core conditions or bounding conditions would be provided.
4. If a new GE fuel or another vendor=s fuel is loaded at the plant, the plant-specific application will reference the fuel-specific stability detect and suppress method supporting the EPU/MELLLA+ operation. The plant-specific application will demonstrate that the analyses and evaluation supporting the stability detect and suppress method are applicable to the fuel loaded in the core.
5. For EPU/MELLLA+ operation, instability is possible in the event of transient or plant maneuvers that place the reactor at high power/low flow condition. Therefore, plants operating at the EPU/MELLLA+ condition must have an NRC reviewed and approved instability detect and suppress method operable. In the event the stability protection method is inoperable, the applicant must employ NRC reviewed and approved backup stability method or must operate the reactor at a condition in which instability is not possible in the event of transient. The licensee will provide technical specification changes that specify the instability method operability requirements for EPU/MELLLA+ operation.

See Reference 30 for RAI response.

<u>Evaluation of RAI 23:</u>
In the RAI response GEH stated the following.

> Per the RAI request, Section 1 of the MELLLA+ LTR will be modified as shown below. Portions of the suggested content of the RAI have been changed to provide consistency with the MELLLA+ LTR and implementation process. For example, each instance of EPU/MELLLA+ contained in the suggested content of the RAI has been changed to MELLLA+. The MELLLA+ LTR is supported by analyses at power levels up to 120% OLTP. However, the LTR is based on the premise that there is no change in power level with the MELLLA+ application. Therefore, the power level for a plant specific application will be the plant=s CLTP, which may not be at the 120% OLTP (EPU) power level.

The RAI response provided the revised introduction Section 1. Revision 2 of the MELLLA+ LTR incorporated the changes.

NRC RAI 24: REACTOR SAFETY PERFORMANCE EVALUATIONS

From the AOO audit, the NRC staff determined that (1) GEH did not provide statistically adequate sensitivity studies that demonstrate the impact of EPU/MELLLA+ operation, (2) [[ ]] (3) the generic anticipatory reactor trip system (ARTS) response may not be applicable for all BWR applications, and (4) the EPU/MELLLA+ impact was not insignificant. The NRC staff also finds that it is not acceptable to make safety findings on two major changes (20 percent uprate based on the CPPU approach and MELLLA+) without reviewing the plant-specific results. Therefore, the NRC staff does not accept GEH=s proposal to [[ ]] EPU/MELLLA+ applications must provide plant-specific fuel thermal margin and AOO evaluations and results. The following discussion summarizes the NRC staff=s bases for concluding that the plant-specific EPU/MELLLA+ application must provide a plant-specific thermal limits assessment and plant-specific transient analyses results.

a. EPU/MELLLA+ Core Design. Operation in the MELLLA+ domain will require significant changes to the BWR core design. Expected changes include (1) adjustments to the pin-wise enrichment distribution to flatten the local power distribution, reduce the r-factor, and increase CPR margin; (2) increased gadolinium (Gd) loading in the bottom of the fuel bundle to reduce the axial power peaking resulting from increased coolant voiding, and (3) changes in the core depletion due to the sequential rod withdrawal/flow increase maneuvers expected during operation in the MELLLA+ flow window. [[ ]] However, the model used for these AOO calculations is not based on a MELLLA+ core, which has been designed for reduced flow at up rated power. Therefore, none of the sensitivity analyses supporting MELLLA+ operation have been performed for a core which includes the unique features of a MELLLA+ core design. Consequently, the effect of MELLLA+ on AOO ΔCPR has not been adequately quantified.

b. Reload-Specific Evaluation of the AOO Fuel Thermal Margin. [[ ]]

c. Off-rated Limits. The NRC staff determined that the off-rated limits (including along the MELLLA+ upper boundary) CPR response may be more limiting than transients initiated from rated conditions. Therefore, AOO results from EPU applications cannot be used as sufficient bases to justify not providing the core and fuel performance results for the plant specific MELLLA+ applications. Moreover, it has not been demonstrated that the generic ARTS limits are applicable and will bound the plant- and core-specific off-rated transient response for all of the BWR fleet. Therefore, off-rated transient analyses must be performed to demonstrate the plant=s ΔCPR response.

d. Mixed Core. Many of the BWRs seeking to implement the EPU/MELLLA+ operating domain may have mixed vendor cores. GEH=s limited (MELLLA+) sensitivity analyses were based on GE14 fuel response of two BWR plants. Additional supporting analyses and a larger MELLLA+ operating experience database will be required before generic conclusions can be reached about the impact of MELLLA+ on core and fuel performance. Specifically, there is no operating experience or corresponding database available for assessing the performance of mixed vendor cores designed for EPU/MELLLA+ operation. As such, plants specific fuel and core performance results must be submitted until a sufficient operating experience and analyses data base is available. In addition, new fuel designs in the future may change the core and fuel performance for the operation at the EPU/MELLLA+ operation. Therefore, the NRC staff=s EPU/MELLLA+ safety finding must be based on plant-specific core and fuel performance.

e. For the CPPU applications, the core and fuel performance assessments are deferred to the reload. Therefore, MELLLA+ LTR proposes that the NRC staff approve an EPU/MELLLA+ application without reviewing the plant=s response for two major operating condition changes. This approach would not meet the agency's safety goals.

See References 18, 28, and 30 for RAI responses.

Evaluation of RAI 24:
GNF stated that the plant-specific EPU/MELLLA+ application will provide plant-specific thermal limits assessment and transient analyses results. The NRC staff accepts this approach.

NRC RAI 25: LARGE BREAK ECCS-LOCA

a. Mixed Core. For a plant-specific EPU/MELLLA+ application, state if equilibrium ECCS-LOCA analyses of each type would be performed or core configuration specific ECCS-LOCA analyses would be performed. If a core configuration specific ECCS-LOCA analyses will be performed, state which NRC-approved codes or methods would be used.

b. Reporting Limiting ECCS-LOCA Results. The MELLLA+ audit indicated that the rated ECCS-LOCA results are reported although it may not be for the most limiting results. For the EPU/MELLLA+ operation, the most limiting ECCS-LOCA result is at the MELLLA+ statepoint of 55 percent CF. Revise the MELLLA+ LTR to state that the ECCS-LOCA result at rated condition, minimum CF at EPU power level and at the 55 percent CF statepoint will be reported. In addition, revise the applicable documents that specify the GEH licensing methods to state that the ECCS-LOCA result corresponding to the rated and the most limiting statepoint will be provided. Report in the supplemental reload licensing report (SRLR), the ECCS-LOCA results at the rated and the most limiting statepoints. Confirm that the steady-state initial conditions (e.g., operating limit maximum CPR [OLMCPR]) assumed in the ECCS-LOCA analyses will be reported in the SRLR.

c. Adder Approach. Was the licensing bases PCT calculated by incorporating a delta PCT adder to the Appendix K PCT? If this is the method used, please justify why the 10 CFR 50.44 insignificant change criteria is acceptable.

See References 20, 29, and 30 for RAI responses.

Evaluation of RAI 25:
RAI 25-a response states that based on the NRC-approved methodology, [[
]] The NRC staff concurs that this is NRC-approved methodology. The codes being used do not model 3D core configuration and therefore the code capabilities do not lend itself for modeling of mixed fuel design ECCS-LOCA core calculation.

The revised RAI 25-b in MFN 05-081 proposes:

1. Calculating the Appendix K and nominal PCT at rated EPU power/flow, rated EPU power and MELLLA+ minimum CF , and the 55% CF MELLLA+ statepoint.

2. Since the MELLLA+ 55% CF would be limiting ECCS-LOCA statepoint for the large break LOCA, the RAI proposes applying off-rated limits at or above the 55% CF statepoint on the MELLLA+ upper boundary. The statepoint above the 55% CF (Point E) statepoint is referred to as E'.

3. The analysis at the minimum CF statepoint (Point D) and E' will be initialized at the rated power linear heat generation rate (LHGR) and the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits. However, for point E', the initial MCPR will include application of the power dependent MCPR multiplier to the rated MCPR.

4. Since credit is taken for the multiplier for those off-rated limits, the plant will be required to apply these limits during the core monitoring systems.

5. These changes will be incorporated in the GEH licensing methodologies and SRLR as follows:

   a) The SAFER/GESTR report will provide the Licensing Basis PCT considering all calculated statepoints. The Licensing Basis PCT will be calculated either using the previous Licensing Basis PCT plant variable uncertainty (e.g., NEDE-23875-1-PA, Section 3.1.3) or with a plant variable uncertainty specific to the calculated statepoint with the highest Appendix K PCT. Only one Licensing Basis PCT will be reported because it is the single PCT, which considers all required licensing conservatism.

   b) Only SRLRs, for both MELLLA+ plants and non-MELLLA+ plants, which report these future SAFER/GESTR analyses will report the Licensing Basis PCT considering all calculated statepoints as described above. No change will be made in SRLR reporting of previous SAFER/GESTR analyses.

   c) Section 6 of NEDC-32950P will be revised to include determining the Licensing Basis PCT considering all calculated statepoints as described above. No other documents that specify the GEH licensing methods will be revised.

   d) The Initial MCPR assumed in the ECCS/LOCA analyses is reported in the SRLR.

In general, the LOCA analyses are performed during implementation of operating changes (e.g., operating domains and EPUs) and during fuel introduction, the analyses are performed using bounding conditions so that cycle-specific LOCA analyses during the reload is not necessary. However, in order to allow operation at the MELLLA+ for plants that are MAPLHGR limited, the RAI proposes the application of the off-rated limits. The NRC staff finds this acceptable provided

the core bundles are monitored based on these multipliers. This will assure that the bundle powers will not be allowed to operate above specific powers that will permit compliance with the off-rated power dependent MCPR limits. In addition, the cycle-specific reload process needs to include confirmation that the ECCS-LOCA off-rated limits are adhered to or it is recalculated if the off-rated limits or the assumed OLMCPR changes. Therefore, the NRC staff accepts the proposal to apply off-rated limits as proposed.

Design-Basis Accident - LOCA

The NRC staff concurs with GEH's proposal. Section 4.3.1 of this SE provides additional discussion and an associated limitation.

Reporting Limiting PCT

Item 5 above addresses the reporting of the limiting ECCS-LOCA PCT response calculated at different statepoints. The approach provided in items are acceptable, with the following changes (1) Both the Licensing and Appendix K PCTs should be reported for all of the calculated statepoints; and (2) The plant-variable and uncertainties currently applied will be used, unless the NRC staff specifically approves a different plant variable uncertainty methods for application to the non-rated statepoints. Section 4.3.1 of this SE provides additional discussion and an associated limitation.

Based on the above discussion, the NRC staff accepts the RAI response. Both this SE and the SE of LTR NEDC-33173P include additional limitations and discussions on the axial power shapes assumed in the ECCS-LOCA analysis and other considerations.

NRC RAI 26: SMALL BREAK ECCS-LOCA RESPONSE

[[

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assuming high pressure coolant injection (HPCI) failure and automatic depressurization system depressurization. At the 55 percent CF statepoint (Point M), the hot bundle may be at a more limiting initial condition in terms of initial void content and the automatic depressurization system (ADS) would depressurize the reactor leading to core uncovery as well. Provide a sensitivity ECCS-LOCA analysis, using the bounding initial condition. Provide a small break LOCA analysis at point M (77.6 percent Power/55 percent CF), based on the bounding initial condition, worst case small break scenario and placing the hot bundle at the most limiting conditions (peaking factors). Use initial SLMCPR and OLMCPR condition that is bounding for operation at 80 percent CF or 55 percent CF statepoint.

See References 29 and 30 for RAI responses.

<u>Evaluation of RAI 26:</u>
The revised RAI 26 response in MFN 05-081 stated:

1. [[

2.

3.

4.

]] of the limiting large break LOCA PCT response, the MELLLA+ plant submittals will include calculations of the limiting small break at rated power/rated CF and rated power/MELLLA+ boundary (point D of Figure 1-1). Discussion of small and large break ECCS-LOCA PCT sensitivity analyses follow:

Small break LOCA

The small break LOCA results provided do show PCT difference of less than [[ ]] between small break LOCA performed at rated and minimum flow MELLLA+ statepoint. The differences between the DBA and the small break LOCA are also less than [[ ]] for small break limited Plant B. However, the RAI response results do not indicate if the reported PCTs are based on Appendix K, the licensing PCT or are nominal. The [[ ]]screening criteria are acceptable if the plant has sufficient margins to the PCT limit of 2200° F. However, for those plants that are LOCA limited, a PCT difference of 20° F can make the difference. Therefore, the margins available need to be included in the screening criteria.

The NRC staff concludes that small break LOCA analysis will be performed for the MELLLA+ minimum CF statepoint for those plants that: (1) are small break LOCA limited for analysis performed at rated EPU conditions; and (2) have margins less or equal to [[ ]]for the Appendix K or the Licensing Basis PCT. For all other plants, the NRC staff accepts GEH's proposed [[ ]]screening criteria. Section 4.3.2.4 of this SE provides additional discussion and an associated limitation.

Table B-8 DBA Limited LOCA PCT

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Table B-9 Small Break LOCA Limited

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NRC RAI 27: SMALL BREAK CONTAINMENT RESPONSE

Using the most limiting small break LOCA, in terms of containment response (possibly at rated condition if limiting), demonstrate whether the suppression pool temperature response to a design basis accident is limiting. Wouldn=t a small break LOCA (e.g., assuming HPCI failure and depressurization of the reactor) be more limiting in terms of suppression pool response? Base your evaluations on the Plant D and Plant E applications.

See References 28 and 30 for RAI responses.

Evaluation of RAI 27:
The intent of the RAI is to establish why small break LOCA which adds energy in the suppression pool yields lower suppression pool temperature relative to DBA, in which break inventory flows into the containment (heat sink).

GEH stated that the peak suppression pool temperature for the small break accident (SBA) with vessel depressurization is not expected to exceed the peak suppression pool temperature for the DBA-LOCA. The RAI response explains the reasons why a DBA will yield higher suppression pool temperature as follows:

1. The key energy sources that affect the peak suppression pool temperature are the vessel decay energy and the initial vessel sensible energy. The decay energy is determined by the decay power time-history and the initial power level. These parameters are the same for both events.
   For a DBA-LOCA, the initial vessel sensible liquid energy is rapidly transferred to the suppression pool during the initial vessel blowdown period. The liquid break flow from

the vessel during the blowdown period partially flashes in the drywell, resulting in a homogeneous mixture of steam and liquid in the drywell. This mixture is forced rapidly from the drywell, through the vent system, to the suppression pool. The vessel is depressurized to the ambient drywell pressure within a few minutes of the start of the event. This effectively transfers the initial vessel liquid sensible energy to the pool within minutes of the start of the event. [[

]] After the vessel blowdown period, relatively cold ECCS liquid from the suppression pool enters the vessel. The ECCS flow floods the vessel to the break elevation and delivers a stream of liquid from the vessel to the drywell. [[

3.

]] After vessel depressurization is completed for the SBA, decay energy continues to produce steam in the vessel. This decay energy is transferred to the suppression pool via intermittent SRV discharges to the suppression pool, which maintains the vessel at low pressure.

4. This process produces a slow heat up of the suppression pool. As with the DBA-LOCA, the peak pool temperature occurs when the energy removal rate by the residual heat removal (RHR) system equals the energy addition rate to the suppression pool. [[

]]

In the RAI response, GEH also performed sensitivity analysis to confirm that the higher PCT is associated with DBA relative to small break LOCA.

The Plant D EPU small break LOCA sensitivity analyses assumed HPCI failure and vessel depressurization. The analyses were performed with: (1) the vessel depressurized with ADS and (2) the SRVs manually controlled and actuated during the vessel. With ADS blowdown, the suppression pool temperature was 204.4° F. The peak suppression pool temperature for the controlled vessel depressurization was 206.9° F. For DBA-LOCA the suppression pool temperature was 207.7° F. The RAI response concludes that for Plant D, the peak small break

LOCA suppression pool temperatures were similar to but not higher than the peak suppression pool temperature for the DBA-LOCA.

The RAI response also cites a SBA analysis performed for the BWR/6-218 plant, assuming manually controlled vessel depressurization. The peak suppression pool temperature obtained from the SBA analysis was slightly higher than the peak DBA-LOCA suppression pool temperature but only by 0.8° F.

GEH concludes that these results confirm that the SBA event does not produce more limiting conditions with respect to peak suppression pool temperature.

Considering that GEH's methodology assumption that the DBA-LOCA always produces the limiting suppression pool, these sensitivity analyses demonstrate that the DBA-LOCA does not necessarily always yield the highest suppression pool temperature. However, the fact that results are close requires consideration. Therefore, the NRC staff concludes that the current methodology is acceptable unless the suppression pool temperature is limiting in terms of containment, equipment performance, environmental equipment qualification or design bases structural analyses (torus attached piping).

NRC RAI 28: ASSUMED AXIAL POWER PROFILE FOR ECCS-LOCA

[[

]] Base your discussion on the predicted response in terms of dry out times. In addition, explain what the axial power peaking would be if the fuel is placed at the LHGR limit at rated conditions, 80 percent CF and 55 percent CF condition. If the axial power peaking would be higher for the non-rated flow conditions, state what axial power peaking were used in the ECCS-LOCA sensitivity analyses reported in MELLLA+ LTR for the 80 percent and 55 percent CF statepoints.

See References 28, 30, and 44 for RAI responses.

<u>Evaluation of RAI 28:</u>

Table B-10 Early Dryout Times for Top and Mid-Peaked Power Profiles

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The above table provides the axial peaking factors used in the analyses supporting the MELLLA+ LTR. [[

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The RAI response also provided the corresponding PCT values as shown in Table B-11. The top-peaked power shape results in slightly higher PCT. The RAI response cites conservative assumptions and concludes that differences are insignificant. The NRC staff confirmatory EPU calculations confirm that the top-peaked power shape is more limiting in the order of 100E F, as would be expected. Therefore, the NRC staff concludes that ECCS-LOCA calculation will be performed with top-peaked power shapes. The NRC staff SE of LTR NEDC-33173P provides additional discussion and an associated limitation.

Table B-11 PCT for Top and Mid-peaked Power Profiles

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				]]

NRC RAI 29: POWER/FLOW MAP

The MELLLA+ LTR states that the slope of the linear upper boundary was derived primarily from reactor operating data. Expand on this statement. Explain what operating data was used. Were all plant types represented? Was the line developed as a bounding line or as a fit to the referred reactor operating data?

See References 28 and 30 for RAI responses.

Evaluation of RAI 29:
The NRC staff finds the response acceptable.

NRC RAI 30: POWER/FLOW MAP

The MELLLA+ minimum statepoint for rated EPU power was limited to 80 percent CF. Explain what the limitations were in establishing the minimum CF statepoint. Similarly, discuss the limitations considered in establishing the 55 percent core statepoint. Discuss why the feedwater heater out-of-service and single loop operation is also not allowed for the EPU/MELLLA+ operation.

See References 28 and 30 for RAI responses.

Evaluation of RAI 30:

The RAI response discussed the predominant factors that influenced establishing the MELLLA+ boundaries. In addition, the RAI response explained why the operational flexibilities such as the FWHOOS or the SLO are prohibited for operation in the MELLLA+ domain.

The NRC staff finds the RAI response acceptable, except that that NRC staff needs more clarification on what is meant by , AFinally, it should also be noted that operation in FWHOOS is considered only a contingency option, for temporary feedwater heater equipment deficiency therefore, this limitation is not expected to impose a significant limitation to plant availability.@ Since the NRC staff review and approval of NEDC-33006P is based on the FWHOOS not allowed due to the higher sub cooling and its impact on stability, any plant-specific application intending to operate with FWHOOS needs to provide the bases in the plant-specific application. Section 9.3.1.3 of this SE provides additional discussion and an associated limitation.

APPENDIX C

NRC STAFF EVALUATION OF PLANT D RAI RESPONSES

TO

SAFETY EVALUATION BY

THE OFFICE OF NUCLEAR REACTOR REGULATION

LICENSING TOPICAL REPORT NEDC-33006P

"GENERAL ELECTRIC BOILING WATER REACTOR MAXIMUM EXTENDED LOAD LINE LIMIT ANALYSIS PLUS"

GENERAL ELECTRIC HITACHI NUCLEAR ENERGY AMERICA, LLC

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APPENDIX C NRC STAFF EVALUATION OF PLANT D RAI RESPONSES

Plant D is a BWR/4 that implemented an extended power uprate (EPU), using the ELTR1/2 methodology with deviation on specific topics. Plant D represents a maximum extended load line limit analysis plus (MELLLA+) pilot plant. The requests for additional information (RAIs) and the associated responses serve as plant-specific examples intended to demonstrate the impact of MELLLA+ operation. They are available in the Agencywide Document Access and Management System in package Accession No. ML072320165. The Appendix C RAI response evaluations also bring specificity to the principal topics of review and highlight the specific areas that the plant-specific MELLLA+ applications should address and focus on.

RAI 1.1: BORON SOLUTION MIXING AND TRANSPORT TIME

The boron mixing and remixing correlations used in ODYN are only a function of mass flow through the jet pumps. Specifically, the boron mixing correlations used in ODYN and TRACG are based on the Vallecitos and Santa Barbara test data. The Plant D ODYN calculations assumed that the SLCS solution injected into the core is at 140° F (enthalpy of 108 BTU/lbm). However, the licensee removed the technical specification (TS) heat-tracing requirement. Therefore, the temperature of the standby liquid control (SLC) solution injected into the core could be as low as 55° F during the winter months.

(1) Evaluate the boron mixing test data and state if the lower solution temperature of 55° F would make the test data inapplicable. A lower boron solution temperature could result in higher stratification of the boron solution in the lower plenum and affect the boron mixing. Explain whether the colder solution temperature would make the boron mixing less effective.

(2) The Plant D ODYN analysis used a boron solution transport time of 30 seconds. A shorter transport time would affect the Plant D anticipated transient without scram (ATWS) response in a non-conservative direction. Explain the basis for the solution transport time used in the Plant D calculation. Include in your discussion the transport time proposed in the NRC-approved ODYN LTR.

Evaluation of RAI 1.1(1):
The NRC staff concurs with the licensee=s evaluation. The density of boron solution changes from 65.3 lb/ft^3 at 140° F to 64.4 lb/ft^3 at 55° F. At nominal conditions, the vessel water density is 47.5 lb/ft^3. Thus, the change in density of the SLC solution at 55° F is negligible and should not affect the boron mixing efficiency assumed in either ODYN or TRACG.

Evaluation of RAI 1.1(2):
The licensee has evaluated the time delay for SLC injection and concluded that it is ~13 seconds for a single pump injection, and ~6.5 seconds for dual pump injection. Thus, the NRC staff agrees with the licensee=s evaluation that the 30-second delay assumed in the ODYN analysis is conservative.

RAI 1.2: STANDBY LIQUID CONTROL SYSTEM (SLCS)

For probabilistic risk assessment (PRA) purposes the licensee-adopted single-pump, single-squib valve success criteria. Operators would continue to initiate both SLC pumps; however, the SLC solution boron-10 enrichment would be such that a single-pump would be able to provide the ATWS shutdown requirement. Plant D single pump squib valve SLC single-pump success criteria assumes a single SLC pump injecting at 43 gpm of boron with a weight-percent concentration of 8.5 and 47 atom percent Boron-10. The MELLLA+ ATWS analyses assume that the hot shutdown boron weight would be injected in less than 20.06

minutes. In addition, the MELLLA+ ATWS analysis assumed two SLC pumps are running, with an injection rate of 66 gpm, using the equivalency values (19.8% natural boron enrichment at 13% concentration).

(a) The MELLLA+ Plant D ATWS Task Report states that the suppression pool temperature and pressure results are bounding as long as the hot shutdown boron weight injection time is less than 20.06 minutes for the chosen SLCS option. Demonstrate that based on the single-pump squib valve criteria and key system parameters, the hot shutdown boron weight can be injected in less than 20.06 minutes.

Evaluation of RAI 1.2:
Hot shutdown boron weight (HSBW) is achieved in ~10.1 minutes if two SLC pumps are available, and in ~19.4 minutes if only one pump/squib valve combination is successful. Thus, the NRC staff concurs that the 20.06 minutes assumption to inject the hot shutdown boron weight.

RAI 1.3: SUPPRESSION POOL TEMPERATURE LIMIT

Identify the [emergency core cooling system – loss-of-offsite accident] ECCS-LOCA containment analysis that is the basis for the 207.7°F ECCS-LOCA suppression pool temperature limit. If this analysis was not based on the MELLLA+ operating conditions, justify why the suppression pool temperature limit of 207.7 F is still applicable. For example, show that the energy deposited into the suppression pool based on the EPU conditions is equivalent or bounding in comparison with the energy deposited into the suppression pool for EPU/MELLLA+ core and operating condition.

Evaluation of RAI 1.3:
The NRC staff agrees that 207.7° F is the Plant D design basis analysis (DBA)-LOCA Suppression Pool Temperature Limit that guarantees containment integrity.

However, the ODYN ATWS calculations have both conservative and non-conservative assumptions, and bear little resemblance to the real ATWS transient. Comparison with sample TRACG best estimate ATWS calculations appears to indicate that the conservative assumptions compensate the non-conservative assumption and the overall ODYN result is likely to be conservative (i.e., yield a larger suppression pool temperature. However, the NRC staff finds it hard to justify the use of the low-fidelity ODYN calculations to make relative comparison between original licensed thermal power (OLTP), EPU, and MELLLA+ ATWS results because many of the most relevant physical effects (e.g., emergency depressurization and the later re-criticality) are not even considered. Thus, the NRC staff does not agree with the licensee=s conclusion that "the long-term suppression pool temperature response does not change with MELLLA+ relative to EPU conditions."

Conclusion:

Therefore, Plant D best-estimate TRACG ATWS calculations are required to evaluate the effect of MELLLA+ operation on suppression pool temperature, vessel overpressure, and peak cladding temperature (PCT). These calculations must include at least main steam isolation valve closure (MSIVC), pressure regulator failed open (PRFO), and loss of offsite power (LOOP) using the values in the plant-specific emergency operating procedures (EOPs), including expected operator actions such as manual safety relief valve (SRV) locking (see response to NRC RAI 2.2). The result of the calculations must at a minimum include the following:

1. Vessel overpressure value

2. Peak PCT for both the early overpressure transient, and during the de-pressurization
3. Peak suppression pool temperature at the point when residual heat removal (RHR) capacity is greater than the core heat generation.

RAI 1.4: SMALL BREAK LOCA AND THE ECCS-LOCA SUPPRESSION POOL TEMPERATURE LIMIT

The generic MELLLA+ topical report proposes [[ ]] In addition, the MLTR also states that the sensible and decay heat do not change with the MELLLA+ operating domain and [[ ]] dispositions the long-term ECCS-LOCA suppression pool heatup evaluation.

(1) Explain why the [[ ]] to establish the suppression pool temperature response. Wouldn't the [[ ]] be more limiting in terms of the suppression pool response for the EPU/MELLLA+ condition? Demonstrate quantitatively based on the EPU/MELLLA+ conditions that the suppression pool heatup for the large break LOCA bounds the suppression pool heatup resulting from a small or intermediate break LOCA with the reactor is depressurized. The evaluation should be based on the MELLLA+ conditions including no [automatic depressurization system (ADS) out-of-service] ADSOOS.

(2) The previous EPU GE14 small-break ECCS-LOCA analysis assumed two ADSOOS (1ADSOOS and 1 ADS single failure). However, footnote to 1 ADSOOS in Section 1.2.4, "Operational Enhancement," states that one ADSOOS applies to "logic only-SRV function must still be available." Explain if this means all ADS valves must be available for the EPU/MELLLA operation. Since two ADSOOS are no longer allowed, explain the effect of all ADS in-service would have on the EPU/MELLLA+ suppression pool heatup.

Evaluation of RAI 1.4(1):
The licensee performed a plant-specific small-break LOCA calculation with ADS for EPU conditions. This calculation shows that the suppression pool only reaches 204° F, as opposed to the 207.7° F for the DBA-LOCA. The MELLLA+ LTR RAI 27 covers the same topic. Note that the 204° F suppression pool temperature is based on ADS blowdown as oppose to manual control of the blowdown. As discussed in the NRC staff's evaluation of RAI 27 (see Appendix B of this SE), the small break LOCA suppression pool temperature can be slightly higher than DBA-LOCA suppression pool temperature. The NRC staff agrees that for LOCA calculations, where the reactor scrams immediately and only decay and coolant sensible heat are of relevance, EPU and MELLLA+ have similar effects on suppression pool temperature; thus the EPU calculation is applicable.

Conclusion:

Both small break and large break could result in consistent or close suppression pool temperature; RAI 27 (Appendix B) provides additional discussion of the impact of small break LOCA and DBA-LOCA on the suppression pool temperature.

Evaluation of RAI 1.4(2):
The licensee makes a distinction between the SRVs being "operable," so they open when the pressure exceeds their setpoint, and having an "inoperable logic," which would prevent manual operation from the control room. On plant-specific bases, the limitation of having 11 SRVs operable is necessary for this application in order to mitigate the peak overpressure early on in

the MSIV isolation event where only automated operation is involved. Thus, the NRC staff agrees with the licensee evaluation. In addition, the licensee has provided a small break LOCA calculation with two SRVs out of service and it resulted in the same suppression pool temperature as with all SRVs in-service, showing that there is no significant effect on the suppression pool.

Conclusion:

The plant-specific application needs to have all 11 SRVs to be operable (i.e., capable of opening due to pressure) for operation within the MELLLA+ operating domain.

RAI 1.5: SUPPRESSION POOL COOLING CAPABILITY

(1) Confirm that the maximum service water temperature is not above 92° F, which was assumed in the ATWS analysis.

(2) RHR TS Operability: The reported Plant D peak suppression pool temperature is based on ODYN and STEMP. The ODYN analysis assumes the water level is maintained at [top of active fuel] TAF+5 and the reactor is not depressurized when the [heat capacity temperature limit] HCTL is reached. The STEMP code is used to calculate the suppression pool temperature. Using the Plant D ODYN SRV flows, the NRC staff finds that the suppression pool temperature is higher than the reported value. This calculation is based on two RHR loops operating. In addition, the evaluation also shows sensitivity to the number of heat exchangers in operation. With one RHR loop in operation, the peak suppression pool temperature is higher than the reported value of 197°F. State whether one RHR loop or two RHR loops in operation are assumed in the ATWS analyses. If two RHR loops are assumed to be operating in the suppression pool cooling mode, then discuss the adequacy of the Plant D TS operability requirement for the RHR system.

(3) The suppression pool cooling capability is important for the ATWS event. The peak suppression pool temperature could be reached after the hot shutdown boron weight is injected into the reactor. The suppression pool temperature could rise until the sensible and decay heat generated is within the RHR suppression pool cooling capability. However, the Plant D long-term ODYN analysis ends when the hot shutdown condition is reached. Please demonstrate why the peak suppression pool temperature would not be reached later in the event (e.g., after the hot shutdown weight is injected.)

(4) Evaluate the RHR system and demonstrate that the high suppression pool temperature would not result in loss of [net positive suction head] NPSH. Include in your evaluation any other design limits that apply to qualification of the RHR system.

Evaluation of RAI 1.5(1):
At Plant D, the service water is supplied from the estuary, which historical has never reached the 92° F limit. The NRC staff accepts the licensee's evaluation.

Evaluation of RAI 1.5(2):
Plant D Technical Specification (TS) 3.6.2.3, "Residual Heat Removal (RHR) Suppression Pool Cooling," requires two RHR suppression pool cooling subsystems operable when in Modes 1, 2, and 3. Thus, the NRC staff agrees with the licensee's evaluation that the two RHR loops need to be operable in order to meet the analysis assumption.

Conclusion:

The TS operability requirement for the RHR system must be consistent with the analysis assumptions, such that two loops of RHR will be required to be operable for implementation of

MELLLA+ if the analysis assumed two operable RHR loops. The plant-specific application should include discussion of the RHR coolant temperature and the number of operable RHR loops assumed operable in the suppression pool temperature analysis. The plant-specific application shall also include the applicable TS changes.

Evaluation of RAI 1.5(3):
The NRC staff disagrees with the licensee evaluation. The licensee states that: (a) "The initiation of emergency blow down effectively achieves hot shutdown of the core. The reactor will remain in a hot shutdown condition until sufficient boron is [injected] to achieve cold shutdown," and (b) "At this point, the RHR heat removal rate would be higher than the heat addition rate and the suppression pool temperature would be decreasing. Hence, the ODYN calculated peak suppression pool temperature without depressurization bounds that from the best estimated TRACG code with depressurization."

With respect to statement (a) above, recent TRACG and NRC confirmatory TRACE calculations, both indicate that the reactor recovers a critical configuration at the end of the emergency depressurization. When re-criticality occurs, occasionally large power spikes (>100% nominal) are observed in some cases. In all cases, the reactor re-pressurizes to an intermediate pressure and remains at significant power for minutes. Since the SRVs remain open, several cycles of de- and re-pressurization are observed for several minutes. Therefore, the reactor does not remain on a hot shutdown condition after emergency de-pressurization as claimed in Plant D. This physical process is not modeled or estimated by the ODYN calculation. For this reason, a limitation is applicable to all plant-specific applications requiring TRACG analysis.

Evaluation of RAI 1.5(4):
The licensee states that AThe NPSH limit for non-accident events (e.g., ATWS) is based on availability of containment overpressure. [[

]] More overpressure credit may be needed for MELLLA+ because of the operation at the higher rodline and the corresponding higher heat load. The suppression pool temperature would be higher relative to the RHR capacity and suppression pool size and capacity.

Conclusion:

As long as the suppression pool remains well below boiling conditions, containment overpressure is minimal. However, many large pumps require significant NPSH to operate. The NRC staff had requested Plant D to evaluate and provide the actual NPSH values for the RHR pumps.

RAI 1.6: LIMITING ATWS STATEPOINT

Evaluation of RAI 1.6:
These review topics were resolved under the generic ATWS analysis RAIs and the content of Revision 2 of the MELLLA+ LTR, NEDC-33006P.

RAI 1.7: Feedwater (FW) Reduction

When the time the FW flow is reduced affects the ATWS and the ATWS instability responses. In general, the ATWS analyses [[

]] Discuss the FW system operation and actuations timing, during the ATWS event. What are the bases for the [[ ]]? At what actuation setpoints or core conditions do the EOPs instruct the operators to reduce the trip or reduce the FW or/and all high pressure systems?

Evaluation of RAI 1.7:
The FW pumps in Plant D are tripped automatically by lack of supply steam. Note: other plant types have a combination of steam and motor operator feedwater pumps, so the pump trip would not necessarily be automatic.

Plant D has turbine driven FW pumps. Following the isolation, the turbines are assumed to continue to operate for [[ ]]

The NRC staff concurs with the licensee's evaluation. The time delays used in the calculations represent the best available approximation of the automated actions of the plant.

RAI 1.8: CONFIRMATION OF ATWS/ATWS INSTABILITY

Section 9.3.3 states that the evaluation of the Plant D ATWS with instability is confirmed to be [[ ]] However, no supporting analysis or evaluation is provided to support this position. Discuss how confirmation was established.

Evaluation of RAI 1.8:
The initial core power-to-flow ratio for Plant D is 51.4 MW/Mlb/hr, which is bounded by the [[ ]] assumed in the generic MELLLA+ LTR. The NRC staff concurs with the licensee that the applicability checklist for generic ATWS/Stability disposition is satisfied for Plant D.

Conclusion:

For plant-specific EPU/MELLLA+ applications, the licensees will evaluate and confirm that the [[ ]] analyses in NEDC-33006P are still applicable. The predominant parameters for the confirmation include any changes in plant design or operation that will result in significantly lower stability margins, such as (1) fuel design changes beyond GE14, (2) the maximum power-flow ratio in the allowed operating domain should not be greater than [[ ]] and (3) any changes in plant design or operation that will increase significantly the subcooling during ATWS events.

RAI 1.9: SRV TOLERANCE AND SUPPRESSION POOL TEMPERATURE

The audit documents indicate that 10 SRVs lift at the TS upper tolerance and one low setpoint SRV lifts at 10% above the nominal lift setpoint. For PRFO at [end of cycle] EOC, assuming one SRVOOS, the limiting Plant D unit reaches a peak ATWS vessel pressure of 1534 psig. Assuming all SRVs are in service, the peak vessel pressure is 1457 psig. However, the integrated SRV flows used to calculate the suppression pool temperatures for all the events are based on an [[ ]]

(1) Explain why a [[ ]] is used for the suppression pool temperature calculations and why a reduced SRV tolerance is used for the peak pressure calculations. The calculated peak suppression pool temperature is 197.7° F, with a proposed suppression pool temperature limit of 207.7° F. In addition, instead of opening and maintaining the SRVs open, the valves are analytically assumed to cycle. Analytically, this assumption could result in an increased the boron mixing and slower suppression pool heatup. Please provide justification for these assumptions. Provide an evaluation of the effects of these assumptions in the calculated suppression pool temperature.

(2) The Plant D audit ATWS calculations show the relief valves' opening duration to be [[ ]] In addition, the relief valve system capacity was determined based on the percent steam flow at 1080 psig. These parameters affect the actual plant ATWS response in terms of pressure relief capability. Explain the basis for these assumed values. Justify why a relief valve capacity based on 1080 psig reactor pressure would be more conservative for peak pressure and suppression pool calculations.

Evaluation of RAI 1.9(1):
The licensee explains that, for ODYN or TRACG analysis, the boron mixing and re-mixing efficiency is determined by an average core flow rate with a [[ ]] Therefore, fast flow oscillations induced by the possibly improper modeling of SRV cycling are averaged out and do not influence the boron mixing efficiency. The licensee states that the 3% Tech Spec tolerance on setpoint drift was used to calculate the initial overpressure transient. The NRC staff agrees with the licensee=s evaluation.

Evaluation of RAI 1.9(2):
The valve characteristics used are based on the generic Target-Rock SRV. The RAI response states that SRV capacity used is the standard nameplate capacity. From previous EPU audits, 90-95% of the valve capacity was used in the analysis. Therefore, it is not clear if the assumption of using the standard nameplate is a change or not. For plant-specific applications, the valve capacities used is flagged as a audit item..

RAI 1.10: HBSW

According to the standard GE methods, ATWS calculations are only performed when new fuel types are introduced. However, the EPGs / SAGS define a procedure to calculate the hot shutdown boron concentration (HSBC), which appears to be cycle-dependent. The calculations specified in the EPG-SAGS procedure include specific control rod patterns and cycle-specific void reactivity coefficients

(1) Calculate the cycle-specific HSBC for the first MELLLA+ cycle at Plant D.

(2) Show that the HSBC value used in the [[ ]] is indeed conservative and applicable to Plant D.

(3) Provide a comparison of the suppression pool temperature response, using the cycle-specific HSBC value and the generic value.

Evaluation of RAI 1.10:
[[

]] In addition, the calculation for HSBW is conservative because it prescribes a no-void conditions. Even under shutdown conditions, decay heat is expected to

provide some level of voiding. Cycle 17 is representative of an equilibrium GE 14 MELLLA+ core.

Table C-1 Plant D HSBW Results

[[

]]

The assumed HSBW is used to determine the timing for the operator to raise the water level to promote boron mixing for lower plenum injection plants. Assuming a higher HSBW delays the time for water level increase and may result in more limiting peak suppression pool temperature. In the reference case, [[

]]

Conclusion:

The NRC staff is not entirely convinced that [[
]]appropriate for all lower plenum injection plants with MELLLA+ core designs. The acceptability of the [[ ]] shall be evaluated on a plant-specific basis.

RAI 2.1: MAXIMUM PRESSURE FOR HEAT CAPACITY TEMPERATURE LIMIT

Section 10.9 of the MELLLA+ topical report states the plant's EOPs will be reviewed for any effects of MELLLA+. The EOPs will be updated, as necessary. Please provide a critical review of the Plant D EOPs for the EPU/MELLLA+ operation. Determine the applicability of the variables, definitions, and calculations specified in the EOP to the EPU/MELLA+ operating condition. The following questions provide some examples of the areas of the ATWS EOP that may need further evaluation and updates in order to determine the limiting values applicable to EPU/MELLLA+ operation.

Maximum Pressure for HCTL Plot (Section 17.5): Section 17.5 defines the procedure for calculating the HCTL. In the example plots (Figs. B-17-5 and B-17-6), a maximum pressure of 1100 psig is used. However, TRACG calculations show that the pressure during MSIV ATWS is consistently above 1100 psig. Please explain whether or not the EPG/SAGs should be modified for EPU/MELLA+ operation to require calculation of the HCTL at the expected higher pressures, and provide the basis.

Evaluation of RAI 2.1:
The NRC staff does not agree with the licensee's statement "the maximum expected reactor pressure is 1130 psig; which is lowest SRV setpoint." While at power, the reactor consistently reaches pressures significantly higher than the SRV lifting pressure. The pressure only decreases to close to 1130 psig when the power is significantly reduced, by either flow reduction, boron injection, or water level reduction. Nevertheless, the NRC staff concurs with the essence of the licensee=s evaluation that suppression temperatures that violate HCTL are not likely to occur at pressures greater than 1100 psig, because the reactor power should have been reduced significantly by then. Therefore, HCTL extrapolation at pressures higher than 1100 psig is not required.

RAI 2.2: HSBW

Evaluation of RAI 2.2:
This topic is evaluated in RAI 1.10 above.

RAI 2.3: BIIT

The BIIT is defined as the suppression pool temperature that will allow for injection of the HSBW without reaching the suppression pool HCTL. Please explain whether or not the BIIT curve should be modified for EPU/MELLLA+ operation.

Evaluation of RAI 2.3:
The BIIT temperature is capped at 100° F, independent of HCTL value. Thus, any changes to HCTL (that are not lower than 100° F) will not affect the BIIT temperature. The NRC staff concurs with the licensee=s evaluation.

RAI 2.4: MINIMUM NUMBER OF SRVS REQUIRED FOR DECAY HEAT REMOVAL

For EPU/MELLLA+, the expected decay heat levels should be higher. Please explain whether or not the minimum number of required SRVs should be changed. Please explain if there are any other variables that would be affected by changing the number of required SRVs.

Evaluation of RAI 2.4:
The minimum number of SRVs required for decay heat removal is based upon the 10-minute decay heat. Operation in the MELLLA+ region will not have a significant effect on the decay heat loading and will not impact the minimum number of SRVs required. The NRC staff concurs with the licensee's evaluation.

RAI 2.5: MINIMUM NUMBER OF SRVS REQUIRED FOR EMERGENCY DEPRESSURIZATION

With EPU/MELLLA+, the expected ATWS power levels should be higher. Please explain whether or not the minimum number of required SRVs should be changed. Explain if there is an effect on any other variables.

Evaluation of RAI 2.5:

The minimum number of SRVs required for emergency depressurization with reactor not shutdown is based on the amount of steam flow through fuel bundles which is required to maintain temperature less than 1500 degrees F. This is a function of the fuel type and not MELLLA+ operation. The NRC staff concurs with the licensee=s evaluation.

RAI 2.6: MINIMUM STEAM COOLING PRESSURE

With EPU/MELLLA+, the expected ATWS power levels should be higher. Please explain if the minimum steam cooling pressure would change. Please explain if there is an effect on any other variables.

Evaluation of RAI 2.6:

The minimum steam cooling pressure is based on the steam flow through the fuel bundle which is required to maintain temperature less than 1500 degrees F. This is a function of the fuel type and not MELLLA+ operation. The NRC staff concurs with the licensee=s evaluation.

RAI 2.7: MINIMUM STEAM COOLING RPV WATER LEVEL

With EPU/MELLLA+, the expected ATWS power levels should be higher. Please explain if the minimum RPV water level would change. Please explain if there is an effect on any other variables.

Evaluation of RAI 2.7:

The minimum steam cooling RPV water level is based upon the fuel type and not expected ATWS power levels. The NRC staff concurs with the licensee's evaluation.

RAI 2.8: MINIMUM ZERO-INJECTION RPV WATER LEVEL

With EPU/MELLLA+, the expected ATWS power levels should be higher. Please explain if the minimum zero-injection RPV water level would change. Please explain if there is an effect on any other variables.

Evaluation of RAI 2.8:

The minimum zero injection RPV water level is based upon the fuel type and not on expected ATWS power levels. This parameter is not used in an ATWS strategy. It is used for steam cooling without injection with the reactor shutdown. The NRC staff concurs with the licensee=s evaluation.

RAI 3.1: EQUIPEMENT OOS

Evaluation of RAI 3.1:
This topic was satisfactorily covered in MELLLA+ LTR and the associated NRC staff safety evaluation and limitations.

RAI 3.2: POWER/FLOW MAP

The principal scoping evaluations, the supporting safety analyses, and the justifications are all based on operation within the MELLLA+ domain, as defined by the power-to-flow equation given in Section 1.2.1 of NEDC-33006P. Any operation outside the MELLLA+ boundary would place the plants in an unanalyzed condition. The following questions focus on the operator training, planned implementation testing, and the TS and operational controls that could provide assurance that the plants could not be inadvertently operated outside the MELLLA+ boundary.

(1) Justify why the MELLLA+ power flow map that specifies the licensed operating domain should not be placed in the TS as specified in 10 CFR 50.36.

(2) Identify the relevant plant operational procedures, training, and software and plant configuration management procedures that need to be updated to implement the MELLLA+ boundary.

(3) Describe any planned testing or surveillance that could be conducted before and during the initial MELLLA+ implementation to ensure that the MELLLA+ boundary is well defined and the units could operate within the MELLLA+ boundary.

(4) Will any planned operator training focus on the challenges of MELLLA+ operation? For example, describe or refer to operator training modules that would cover instability performance, ATWS, and the bases for the MELLLA+ restrictions (e.g., FWHOOS, FFWTR, SLO, 1ADSOOS, 1SRVOOS) or the potential instability performance in the event of an RPT or turbine trip.

(5) Describe the updates that would be made to the online monitoring system, including the process computer and core monitoring packages such as 3D MONICORE and Powerplex. The NRC staff is interested in how the MELLLA+ operating domain will be defined in the process computer and in the core monitoring programs used to predict the plant performance before any plant evolution.

(6) There is a potential that plant maneuvers from within the licensed domain may cause the reactor to operate outside the MELLLA+ operating domain. Since EPU/MELLLA+ conditions have reduced the plant s available margin in the safety analyses, operation outside the MELLLA+ boundary is not permissible.

1. Describe the steps that will be taken or the tools will be used to analytically predict where the plant's response will be during power maneuvers, accounting for the rod patterns and specific core conditions?

2. Define the cycle-specific 100% loadline.

3. The NRC staff understands that the actual plant operating loadline varies within a cycle and from cycle to cycle. For example, the cycle-specific load can change depending on the change in the feedwater temperature as a function of power. Discuss how and why the actual plant-specific loadline is expected to vary throughout the cycle and from cycle to cycle. Explain how it will be assured that the plant's cycle-specific 100% rodline will not exceed the MELLLA+ domain. What reportable requirements will be in place to inform the Commission if the plant is operated outside the MELLLA+ domain, including operation outside the domain at the off-rated power levels.

Evaluation of RAI 3.2:

The RAI response provided the requested discussion and evaluation; and is therefore acceptable.

RAI 4.1: ECCS COOLING PERFORMANCE AND SMALL BREAK LOCA RESPONSE

Evaluation of RAI 4.1:
This topic was satisfactorily covered in a similar generic RAI response and the associated staff evaluations and limitations.

RAI 4.2: LARGE BREAK LOCA

Evaluation of RAI 4.2:

This topic was satisfactorily covered in a similar generic RAI response and the associated staff evaluations and limitations.

RAI 4.3: MAPLHGR AND MCPR MULTIPLIERS

Evaluation of RAI 4.3:

This topic was satisfactorily covered in the MELLLA+ LTR and the associated staff safety evaluation and limitations.

RAI 5.1: STABILITY BACKUP STABILITY PROTECTION

Evaluation of RAI 5.1:

This topic was satisfactorily covered in the MELLLA+ and DSS-CD LTRs and the associated NRC staff safety evaluations and limitations.

RAI 5.2: DSS-CD TECH SPEC CHANGES

Evaluation of RAI 5.2:

This topic was satisfactorily covered in the MELLLA+ and DSS-CD LTRs and the associated NRC staff safety evaluations and limitations.

RAI 6.1: HPCI AND RCIC PERFORMANCE

Provide the HPCI and RCIC maximum design pressures and explain if these systems can inject into the reactor throughout the transient event. For example, with the reactor still pressurized, can the HPCI and RCIC systems inject and maintain the EOP-defined ATWS water level?

Evaluation of RAI 6.1:
The licensee reviewed the performance of HPCI and RCIC during the 105% power uprate, where the SRV setpoints were increased by 25%. They found the performance adequate as long as the pressure stayed below 1164 psig, the SRV lifting pressure. TRACG simulations indicate that, except for a short period of time early in the transient, the vessel pressure remains at or below ~8 Mpa (~1175 psig). Thus, the NRC staff concurs with the licensee evaluation that HPCI and RCIC provide sufficient pressure for water injection during ATWS events.

RAI 6.2: HPCI WATER SOURCES

Although the CST is the preferred water source, the suppression pool is the safety water source system. Will HPCI system automatically switch to take suction from the suppression pool, when the suppression pool water level high condition is reached? If so, explain whether the suppression pool heatup during an ATWS event would affect the HPCI and RCIC operability and qualification.

Evaluation of RAI 6.2:

The licensee states that HPCI is designed for continuous operation at a temperature of 140°F or lower. Operators are instructed to switch HPCI source to CST if the suppression pool temperature reaches this limit. CST is a non-safety grade source of water, but is expected to have sufficient inventory to ride an ATWS event - it provides for approximately one hour of HPCI inventory at hot shutdown. The licensee states that the use of a non-safety grade source of water for HPCI is acceptable during ATWS.

For this stand-pipe injection plant, the NRC staff concurs that the use of non-safety grade CST water is acceptable for ATWS event, if suppression pool temperature is high and NPSH head credit is not a solution. Specially, the HPCI NSPH is limited to 140°F, because of pump oil system. Therefore, containment overpressure does not appear as a solution, since at issue is not the pump cavitation protection only. For suppression pool temperature greater than 140 °F, the HPCI operability appears to be tied to the duration that the CST water inventory would last.

In general, the acceptability of HPCI water sources, during ATWS shall be evaluated on a plant-specific basis.

RAI 6.3: NPSH LIMITS DURING ATWS

Similarly, when increasing the water level after depressurization of the reactor, does the Plant D ATWS EOP provide actions to take when the suppression pool temperature is high in terms of NPSH? Please provide a discussion to demonstrate that the systems would be able to provide the core cooling and coverage throughout an ATWS event in terms of the NPSH of the high and low pressure systems.

Evaluation of RAI 6.3:
The licensee states that the plant-specific HCTL is between 160 °F and 168 °F for the conditions expected. HPCI can operate with suction water up to 170 °F for short periods of time.

Therefore, HPCI will be available prior to emergency depressurization with source water from either suppression pool or CST. Following depressurization, low-pressure systems can be used to restore level.

The RAI response did not cover why the high suppression pool temperature will not affect the NSPH requirements for the low pressure ECCS systems. From the plant-specific evaluations, the NRC staff finds that plants will be limited by the availability and operability of the safety system.

As discussed in RAI 6.2, the water level control before and during the depressurization phase will depend on the suppression pool temperature remaining below 140 °F, which is a limit that applies to the HPCI pump irrespective of the available overpressure. The CST water inventory may not be enough to ensure HCPI operability for the duration required.

The NRC staff finds detail integrated system evaluations need to be performed to ensure that all the systems will be available, when required and assumed, considering the suppression pool temperature with time in the duration of the event, and the CST water level inventory. As discussed in the MELLLA+ SE, early hot shutdown through high B-10 concentration would support reducing the suppression pool temperature and ensuring system availability.

In general, the acceptability of water sources for high and low pressure systems, during an ATWS shall be evaluated on a plant-specific basis.

RAI 7.1: SPENT FUEL CRITICALITY

There is no spent fuel criticality evaluation in the EPU/MELLLA+ application. For the EPU/MELLLA+ operation, fuel with higher plutonium content and larger batch fractions may be placed in the core. Provide an evaluation that demonstrates that the spent fuel criticality analysis-of-record will be applicable and bounding for spent fuel loading patterns and conditions expected to exist at the pool with continued operation under the EPU/MELLLA+ condition.

Evaluation of RAI 7.1:
The RAI response explained the requirements and process that ensures the spent fuel criticality is mitigated and the TS requirements are met. The spent fuel pool criticality acceptability is determined prior to the fuel order. The RAI response did not specifically address the impact of increased Pu content or how changes in how the fuel is operated are included in the spent fuel criticality analysis. In addition, since EPU core designs involve high batch fraction (~40% or higher, the impact of higher discharge bundles per cycle on the spent fuel pool capacity also needs to be assessed.

Assuming that the burnup effects corresponding to the projected M+ operation is properly accounted for, the NRC staff finds the response acceptable. However, the plant-specific application should include confirmation or discussion on how the spent fuel criticality requirement can be met for bundles that operated at MELLLA+ conditions.

RAI 8.1: ODYN CALCULATION WITH ALL SRVS IN SERVICE

GE provided the Plant D ODYN run with one SRVOOS. Provide the Plant D ODYN run with all SRVs in service. This is important, because: (1) the ODYN run stops after hot shutdown boron weight is injected, and (2) the SRV flow would be higher with all SRVs in service.

Evaluation of RAI 8.1:
The data was not provided. The licensee states that the number of SRVs in service does not significantly affect the final suppression pool temperature because SRV cycling does not enhance boron mixing in the ODYN model. In additional the ultimate energy deposited into the containment is dependent on reactor power. A plant-specific evaluation of this equipment option will be performed.

RAI 8.2: PEAK SUPPRESSION POOL TEMPERATURE

The peak suppression pool temperature can occur after the hot shutdown weight is injected into the reactor. Decay and stored energy would continue to be added into the reactor until the heat rejected into the suppression pool is within the capability of the RHR suppression pool cooling capability. Therefore, provide ODYN SRV flows data and STEMP suppression pool temperature calculations that extend the analyses until the suppression pool temperature reaches equilibrium condition or is decreasing.

Evaluation of RAI 8.2:
The requested information was provided and is acceptable to the NRC staff.

RAI 8.3: TRACG Analysis Detailed Data

The preliminary TRACG analysis the NRC staff reviewed during the audit did not include peak suppression pool temperature calculation. Instead, GE compared the integrated SRV flows between the ODYN analysis (based on TAF+5) and the TRACG results (based on TAF+5 and TAF). The TRACG sensitivity analyses modeled reactor depressurization. However, during the audit the TRACG cases were experiencing problems after the water level is raised. In addition, the audit TRACG results showed significant difference between the PCT after depressurization and the reported ODYN PCT. The reported ODYN PCT is the PCT during the pressurization phase, since ODYN does not model depressurization. In subsequent discussion, GE stated that the TRACG analyses problems were resolved and the TRACG cases can simulate plant response throughout the ATWS event. In addition, GE reported that by modeling in the radiation heat transfer, the TRACG depressurization PCT is within the reported ODYN pressurization PCT results. Using the TRACG depressurization sensitivity analyses,

(1) Provide the SRV flow data. Extend the ATWS analysis until the suppression pool temperature reaches equilibrium or is decreasing. State if one or two RHR loop are assume to be in operation. Include the results from the comparative TAF+5 ODYN results.

(2) Provide STEMP suppression pool temperature results and the comparative ODYN TAF+5 STEMP results.

(3) Provide the TRACG ATWS input files. Provide the TRACG sensitivity analyses output files.

(4) Please provide documentation of all of the changes made to the audit TRACG analyses that reduced the PCT values after the depressurization (e.g., including radiation heat transfer). Provide a brief description of how the code problems associated with increasing the water level after depressurization were resolved. Note that the code is not approved for modeling of all of the ATWS events.

(5) For the TRACG sensitivity analyses, determine if the mitigating system (e.g., HPCI) can perform the analytical assumed function, considering the suppression pool temperature. If the suppression pool condition during the event is beyond the system operability requirement, justify the bases for the analysis assumption.

Evaluation of RAI 8.3:
The licensee states that for ATWS evaluations, all equipment not involved in the event initiation is assumed to be operational except for the control rod insertion. Thus, both RHR loops are active for this analysis. The long term RHR efficiency is determined by the STEMP code, not ODYN. STEMP has a built in decay heat model that is input to the suppression pool heat up rate; therefore once ODYN reaches hot shutdown, STEMP continues the calculation.

The RAI response provided summary of the TRACG changes and is acceptable to the NRC staff.