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 <u>NRC STAFF EVALUATION OF FUEL DEPENDENT ANALYSES RAI RESPONSES</u>

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.

<u>Evaluation of RAI 1:</u>
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:

(1) [[

(2)

(3)

]]

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: [[

]]

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.

<u>Evaluation of RAI 6:</u>

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.

<u>Evaluation of RAI 7:</u>
The RAI response provided that the active coolant average void fraction, excluding the unheated and bypass regions.

$$< VF > = \frac{\sum_{i=1}^{\#\,each\,type} n_i \frac{\sum_{k=1} VF_k FlowArea_k}{24 < FlowArea >}}{Total\ \#\ of\ Bundles}$$

, 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 [[
]]
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. [[

]]

See References 30 and 32 for RAI responses.

<u>Evaluation of RAI 10:</u>
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.

<u>Evaluation of RAI 12:</u>
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.

<u>Evaluation of RAI 13:</u>
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.

<u>Evaluation of RAI 14:</u>

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.

<u>Evaluation of RAI 15:</u>
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.

<u>Evaluation of RAI 16:</u>
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.

<u>Evaluation of RAI 17:</u>
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.

<u>Evaluation of RAI 18:</u>
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.

<u>Evaluation of RAI 19:</u>
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.

<u>Evaluation of RAI 20:</u>
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.

<u>Evaluation of RAI 21:</u>

The RAI response stated that [[

]]

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.

<u>Evaluation of RAI 27:</u>
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.
2. 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. [[

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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

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]] 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 100° 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 , "Finally, 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.