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

TABLE OF CONTENTS

LIST OF TABLES

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.

<u>Evaluation of RAI 1.1(1):</u>
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.

<u>Evaluation of RAI 1.1(2):</u>
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.

<u>Evaluation of RAI 1.2:</u>
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.

<u>Evaluation of RAI 1.3:</u>
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 “The 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.

<u>Evaluation of RAI 3.2:</u>

The RAI response provided the requested discussion and evaluation; and is therefore acceptable.

RAI 4.1: ECCS COOLING PERFORMANCE AND SMALL BREAK LOCA RESPONSE

<u>Evaluation of RAI 4.1:</u>
This topic was satisfactorily covered in a similar generic RAI response and the associated staff evaluations and limitations.

RAI 4.2: LARGE BREAK LOCA

<u>Evaluation of RAI 4.2:</u>

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.

<u>Evaluation of RAI 6.3:</u>
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.