APPENDIX A

NRC STAFF EVALUATION OF ATWS RAI RESPONSES

TO

SAFETY EVALUATION BY

THE OFFICE OF NUCLEAR REACTOR REGULATION

LICENSING TOPICAL REPORT NEDC-33006P

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

GENERAL ELECTRIC HITACHI NUCLEAR ENERGY AMERICA, LLC

TABLE OF CONTENTS

LIST OF FIGURES

LIST OF TABLES

APPENDIX A <u>ATWS RAI EVALUATIONS</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 20, 29, and 30.

NRC RAI 1: LOOP EVENT DISPOSITION

Discuss how it will be determined [[

]]

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

]].

See Reference 31 RAI I-1.1 for RAI response.

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

]]

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

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

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

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

See Reference 31 RAI I-1.2 for RAI response.

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

NRC RAI 3: DETERMINING PEAK CLADDING TEMPERATURE (PCT)

RAI 3-1: Initial Hot Bundle Operating Conditon

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

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

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

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

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

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

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

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

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

RAI 3-2: Determining PCT

Provide a table showing the previous PCT results used to make the assessment. List the MELLLA+ PCT sensitivity analyses the MLTR is referring to. Describe the key assumptions used for the PCT calculations (BWR type, fuel type, rodline and power level, etc.). Identify if ODYN/ISCOR/TASC combination or TRACG was used in calculating the PCT.

See Reference 31 RAI I-2.2 for RAI response.

Evaluation RAI 3-2:
This RAI was based on the initial Revision 1 of NEDC-33006P, which proposed not providing the ATWS PCT calculation.

The ODYN/ISCOR/TASC results were provided. The PCT values are well within the 2200° F limits.

However, the NRC staff expects all BWRs would require depressurization during an ATWS event under EPU/MELLLA+ conditions. Therefore, there are two significant PCT events: Initial over-pressurization phase, and during the emergency depressurization phase. Depending on the water level strategy followed by the plant, the PCT event for the second phase could be higher than for the first phase. Both PCT phases will be captured by the plant specific TRACG ATWS calculations. Revision 2 to the LTR, updated the ATWS section and includes the commitment to provide plant-specific TRACG PCT calculation.

In addition, Table A-1 shows PCT values using ODYN/ISCOR/TASC calculation in which the PCT goes down for the EPU/MELLLA conditions. GEH/GNF-A attributes the lower PCT response for 20% higher power to flow re-distribution because the average channel power increases while the hot channel is assumed to operate the maximum design limit. However, the NRC staff believes that the modeling limitation of the ODYN/ISCOR/TASC code system is contributing to the lower PCT response. The code system does not represent the flatter radial power distribution of the EPU cores, and the distribution of large fraction of high powered bundles in the core. Therefore, the flow redistribution with in the average and the hot bundles modeled in ODYN/ISCOR/TASC may not capture the actual bundle flow distribution of an EPU core, with high batch fraction, more high powered bundles.

Note that the ATWS analysis is performed at the minimum core flow statepoint, which is considered to be more limiting, because the RPT is less effective in adding negative void reactivity relative to ATWS initiated from rated conditions. Therefore, all these calculations are performed at higher rodline than rated OLTP conditions. For example, the MELLLA/OLTP statepoint at 75% core flow statepoint rodline corresponds to higher rodline than the rated OLTP 100% rodline. Therefore, for MELLLA/ 75% core flow and MELLLA/EPU, the reactor may reach similar power levels after RPT. However, as can be seen in Figure 3-2 below, after MSIV closure, the peak neutron flux initiated from MELLLA+ at 85% core is much higher than MELLLA at OLTP/75% core flow statepoint. Therefore, since the initial peak PCT corresponds to the initial power peak due to the MSIVC, it is not clear why the MELLLA+ PCT is lower,

However, with the plant-specific TRACG sensitivity analyses, the core thermal hydraulic conditions will be modeled more accurately.

Table A-1 PCT Comparisons at ELLLA/OLTP, MELLLA/OLTP &105%P, MELLLA/EPU and MELLLA+/EPU

Peak Cladding Temperature (ºF)
(All Calculations Based on ODYN/ISCOR/TASC Methodology)

[[					

]]

[[

]]

Figure A-1 Neutron Flux response for MSIVC ATWS Event.

NRC RAI 4: ODYN ATWS CALCULATIONS

The NRC staff has reviewed ODYN data for ATWS events for three plants (Plant D, Plant E, and Plant F) at two operating conditions (100 percent OLTP, 75 percent flow, and 120 percent OLTP, 85 percent flow). The following RAIs address the key assumptions and system actuations used for these analyses.

RAI 4-1: Plant D ATWS Response

The Plant D MELLLA+ LTR (NEDC-33063P) states that the peak vessel pressure for an MSIV isolation ATWS is 1457 psig. [[

]]

1. What is the difference between the two calculations?
2. What is the applicable peak pressure limit?
3. If the applicable limit is 1500 psig, is it violated by the ODYN calculation results provided?
4. Give what the peak pressures are for other analyzed ATWS cases, including PRFO.
5. Provide the ODYN results as a function of time for the limiting ATWS event for Brunswick.

See Reference 31 RAI IV-1 for RAI response.

<u>Evaluation RAI 4-1:</u>
The main difference among the various calculations is the number of SRVs assumed out of service. The upgrade to MELLLA+ conditions mandate that all 11 SRVs be operational to satisfy the 1500 psig over-pressure limit for Brunswick. If one SRV is placed out of service, the plant must exit the MELLLA+ region. Table A-2 shows the peak pressure values corresponding to different cycle exposures and with one or no SRVOOS.

Table A-2 Plant D Peak Vessel Pressure at MELLLA+

Event	Cycle Exposure	Number of SRV OOS	Peak Vessel Pressure (psig/sec)
MSIVC	BOC	[[	
MSIVC	EOC		
PRFO	BOC		
PRFO	EOC		
PRFO	EOC		]]

The NRC staff concludes that MELLLA+ operation has a significant effect on the initial over-pressure following isolation events. Note that the TS SRV LCO is attributed to the ASME Overpressure transient analysis. However, the SRVs are relied upon in meeting the ATWS acceptance criteria. As can be seen by this example, the number of SRVOOS affects the plants' capabilities to meet the ATWS acceptance criteria. Therefore, the allowed TS SRVOOS need to be consistent with the number assumed in the ATWS analysis. Section 9.3.1.3 of this SE provides additional discussion and an associated limitation.

RAI 4-2: Timing of Operator Actions

It is customary in safety calculations to allow some time for operator actions. It is apparent from a review of the ODYN results that operator actions occur in very short timeframes. [[

]] Explain the assumptions used for operator actions during these analyses.

See Reference 31 RAI IV-2 for RAI response.

Evaluation RAI 4-2:
For the ODYN calculations, operator actions assume a 2 minute delay. Some ATWS mitigation actions are automatic and, for those, plant-specific settings are used. [[

]]

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

RAI 4-3: ODYN Calculation Assumptions

In the Plant D calculation, the water level is raised at exactly 1400 seconds (in the Plant F calculation at exactly 1600 seconds). According to the EPGs, the water level is supposed to be raised when the HSBW has been injected into the core. What is the basis for the exact [[ ]] used? Shouldn't the time when the HSBW is reached be dependent on the SLC injection initiation time?

See Reference 31 RAI IV-3 for RAI response.

Evaluation RAI 4-3:
The water level in the Plant D calculation is [[

]] Therefore, the NRC staff agrees that the ODYN calculations represent the expected timing of operator actions.

RAI 4-4: Limiting ATWS Events and Scenarios

Provide the assumptions used in the ATWS analysis for the EPU/MELLLA+ pilot plant [Plant D] specific calculation (NEDC-33063P). Specifically, what type of ATWS transient is limiting? What are the initial conditions, including the power, flow, and the suppression pool level? What operator actions are assumed? What ATWS mitigation actions are implemented during the transient? What values are used for EOP variables (e.g., HCTL, HSBW, etc.)?

See Reference 31 RAI V-3 for RAI response.

Evaluation RAI 4-4:
Table A-3 shows the requested information, except the EOP variables of HCTL.

Table A-3 Plant D ATWS Analysis Assumptions –

Bounding ATWS events	MSIVC and PRFO
Initial Reactor Power (% OLTP)	120

Initial Core Flow (% NBR Rated)	85
Initial Suppression Pool Mass (lbm)	5,365,000 at minimum water level
Operator Actions Assumed	1. Maintain water level at TAF+5' following feedwater pump trip 2. [[ ]] 3. Increase water level after HSBW injected
ATWS Mitigation Implemented	1. Reduce water level to TAF+5' 2. Inject boron
EOP Variable Used	[[ ]]

NRC RAI 5: IMPACT OF MELLLA+ ON THE HSBW VALUE

Because MELLLA+ operation occurs at a higher control rod line, one would expect the HSBW to increase over the baseline. The analysis assumed the same HSBW value for both MELLLA+ and the previous baseline condition. Under MELLLA+, the HSBW may be higher, leading to a longer time of suppression pool heating before the water level is raised to remix the boron at the bottom of the vessel, which achieves the hot shutdown condition. What is the effect of using a MELLLA+ specific HSBW value on ultimate suppression pool temperature?

See Reference 31 RAI IV-4 for RAI response.

Evaluation RAI 5:
A sensitivity analysis was provided by GEH in the response to this RAI. Table A-4 below provides the sensitivity analysis for all conditions, the ODYN criteria was satisfied; however, using more realistic TRACG analysis, the results show that the suppression pool HCTL will be exceeded and emergency depressurization would be necessary.

Table A-4 Plant D HSBW Sensitivity Results

HSBW (ppm)	HSBW Injection Time (sec)	Level Increase Time (sec)[1]	Approximate Hot Shutdown Time (sec)[2]	Peak Pool Temp (F)
[[				
				]]

[1] The level increase time is the sum of HSBW injection time, SLCS initiation time, and Boron transport time.
[2] [[
]]

NRC RAI 6: SRV INDUCED FLOW OSCILLATIONS

The EPGs instruct the operator that a number of SRVs should be locked open to prevent cycling (and prevent possible mechanical failures). By allowing the SRVs to cycle, the core flow oscillates wildly because of the SRV-induced pressure transients. By increasing the flow values over the non-mixing stagnation flow value in the boron correlation, these wild flow oscillations promote Boron mixing that otherwise would not happen. Explain why it is conservative to allow these wild flow oscillations to continue, thus increasing the amount of boron mixed with the core inlet coolant and reducing the reactor power.

See Reference 31 RAI IV-5 for RAI response.

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

]] Therefore, even though the ODYN calculation does not accurately represent the SRV open/close cycles expected during the early phase of the ATWS event in the real event, boron mixing is not enhanced by this fluctuations, and the results are representative of real conditions.

NRC RAI 7: SRV TOLERANCES USED

Section 9.3.1 of the Plant D MELLLA+ LTR (NEDC-33063P) states that the MELLLA+ analysis was performed with 10 percent SRV tolerance, rather than the normally assumed 3 percent tolerance. Provide an explanation of the detailed SRV lifting pressures (including the tolerance) and the percent of nameplate flow used for the calculations.

See Reference 31 RAI IV-6 for RAI response.

<u>Evaluation RAI 7:</u>
The Plant D ATWS overpressure evaluation was performed with the lowest opening SRV tolerance of 10%. All other 10 groups use the standard 3% tolerance. However, the RAI response does not discuss whether the 10% tolerance was used to bound potential high valve drifts of the lowest opening SRVs. In addition, the RAI response states that SRV nameplate capacity is 829,000 lbm/hr, based on reference pressure of 1080 psig. Further discussion is required on the SRVs capacity through out the event scenario.

Recent experience indicates that some licensees have used valve tolerances that are less than the actual valve performance, even though LERs had been issued on the results of valve testing data outside Tech Spec ranges. The valve tolerance uncertainty treatment should be based on a NRC approved method or the historically recommended NRC method of 95/95 or plant-specific setpoint methodology needs to be reviewed and approved.

NRC RAI 8: ATWS SEQUENCE OF EVENTS

Provide the sequence of events (including SLC injection and water level reduction times) for these calculations. Specify the actuation setpoints and initiation times. What are they based on?

See Reference 31 RAI IV-7 for RAI response.

<u>Evaluation RAI 8:</u>
The sequence of events for Plant D with for the one SRVOOS was provided. Note that the peak pressure with 1 SRVOOS exceeded the ATWS pressure limit of 1500 psig.

Table A-5 Sequence of Events for an MSIVC Event

Item	Response	M+ Event Time (sec)
1	MSIV Isolation Initiates	[[
2	High Pressure ATWS Setpoint	
3	MSIVs Closed	
4	Peak Neutron Flux	
5	Opening of the First Relief Valve Tripped	
6	Recirculation Pumps Tripped	
7	Peak Heat Flux Occurs	
8	Peak Vessel Pressure	
9	Feedwater Reduction Initiated	
10	BIIT Reached	
11	PCT Occurs	
12	SLCS Pumps Start	
13	Water Level Increased	
14	Hot Shutdown Achieved [[ ]]	
15	Peak Suppression Pool Temperature	]]

Table A-6 Sequence of Events for an PRFO Event

Item	Response-	M+ Event Time (sec)
1	Turbine Control and Bypass Valves Start Open	[[
2	MSIV Closure Initiated by Low Steamline Pressure	
3	MSIVs Fully Closed	
4	Peak Neutron Flux	
5	High Pressure ATWS Setpoint Tripped	
6	Opening of the First Relief Valve Tripped	
7	Recirculation Pumps Tripped	
8	Peak Heat Flux Occurs	
9	Peak Vessel Pressure	
10	Feedwater Pumps Runback Initiated	
11	BIIT Reached	
12	PCT Occurs	
13	SLCS Pumps Start	
14	Water Level Increased	
15	Hot Shutdown Achieved [[ ]]	
16	Peak Suppression Pool Temperature	]]

Table A-7 Initial Valve Opening Timing for an MSIVC Event and a PRFO Event

Valve Group	MSIVC Event	PRFO Event
1	[[	
2		
3		
4		
5		
6		
7		
8		
9		
10		
11		]]

NRC RAI 9: PCT DISPOSITION AND FUTURE FUEL DESIGNS

Justify why the sensitivity results, based on performance of GE fuel (up to GE14), form the bases for [[

]] Alternatively, state that the coolable geometry (e.g., PCT) and the 17 percent local cladding oxidation acceptance limit for the ATWS analyses would be demonstrated on a plant-specific basis, if another vendor's fuel, new GNF fuel, or mixed vendor cores are involved. In the latter case, revise the MLTR and include a specific applicability statement.

See References 20 and 31 RAI I-2.3 for RAI response.

<u>Evaluation of RAI 9:</u>
This RAI is based on Revision 1 of NEDC-33006P. The NRC staff disagrees with the evaluation. Even though large margin has been demonstrated for the lead plant, the consequences of failure of this criterion are large. Plant specific calculations will guarantee that the criterion is met. See details in section NRC RAI I-3.3 below.

Since licensee's can introduce new fuel without explicit NRC approval, it is not clear what ATWS analysis would be performed during the new fuel introduction phase. The plant-specific application would include TRACG ATWS analysis. In evaluating the TRACG sensitivity analysis, the NRC staff would determine on plant-specific bases, if additional limitation is required. This is especially relevant in the applications showing that the plant is limited in terms of PCT.

Revision 2 of the LTR addresses the NRC staff concerns and no limitation is needed.

NRC RAI 10: LIMITING ATWS RESPONSE AND POWER/FLOW STATEPOINT

Explain why the ATWS analysis performed at the minimum core flow statepoint is more limiting than the analysis performed at the maximum achievable core statepoint for the EPU/MELLLA+ operation.

See References 29 and 31 RAI I-2.4 for RAI response.

<u>Evaluation RAI 10:</u>
EPU experience indicates that BWRs are limited by the ATWS peak pressure. Consequently, the sensitivity if the ATWS pressurization response to the initial flow statepoint is important. In the RAI response, GNF-A states that the PCT and the suppression pool response would be more limiting from the minimum core flow statepoint rather than at the rated core flow statepoint.

The power level during the ATWS event (after flow reduction) is controlled mostly by the operating control rod line. The low-flow initial condition at rated power is in a higher control rod line, so the power after the flow reduction will be larger. Although the control rod line is the dominant effect; other mechanism (like flow redistribution) are in play, could result in second order perturbations. This combined with the assumed radial peaking factor at the low flow condition and the higher void conditions could lead to higher PCT. Similar to the ASME overpressure response, it is feasible that the peak ATWS response could be more limiting at the rated or maximum core flow statepoints, depending on the control rod pattern assumed and the exposure. However, considering that the ATWS RPT is less effective if initiated from the minimum flow statepoint (80%) compared to the rated core flow statepoint and the that the minimum core flow is at ~ 140 % rodline, the NRC staff accepts that the minimum core flow is the limiting statepoint for the ATWS analysis.

NRC RAI 11: APPLICABILITY OF THE ODYN LICENSING METHODOLOGY TO THE MELLLA+ ATWS

The Emergency Procedure Guidelines (EPGs) require a number of operator actions, and they allow a range of water level control strategies during isolation ATWS events, from 2 feet below the feedwater spargers to the minimum steam cooling water level (MSCWL). However, limitations in the approved ODYN methodology only allows for an ATWS calculation with a minimum water level of top-of-active (TAF+5 ft), and do not allow for accurate modeling of all required operator actions (such as depressurization when the heat capacity temperature limit (HCTL) is reached). The relevant question is whether the approved ODYN ATWS methodology provides conservative results that can be used to evaluate the impact of MELLLA+ operation on ATWS performance.

RAI 11-1: ODYN Limitations

Provide a description of the approved ODYN ATWS methodology and its limiting assumptions (e.g., control level at TAF+5, do not depressurize). Provide a description of the treatment of uncertainties in approved ODYN licensing calculations.

See References 29 and 31 RAI I-3.1 for RAI response.

<u>Evaluation RAI 11-1:</u>
The RAI response provided description of ODYN limitations for ATWS analysis as follows:

1. The downcomer level must remain above the jet pump suction and no prolonged level in the active channel is allowed:
2. The duration of the simulation after the upper plenum subcools should be limited.
3. The mass in the separators should not remain zero and, therefore, the code is restricted to applications where the water level remains at or above the top of active fuel plus 5 feet.
4. The code is not presently qualified to perform stability calculations.
5. No lower plenum voiding is allowed.

The NRC staff agrees with these ODYN limitations. However, the response did not address the following part of the RAI. “Provide a description of the treatment of uncertainties in approved ODYN licensing calculations.” Does ODYN cover how valve tolerances/uncertainties are going to be addressed?

In MFN 05-081, GEH provided a revised RAI response addressed the uncertainties and valve tolerances assumed in ODYN. The RAI response refers to Section 5.6 of NEDC-24154P-A, (Supplement 1-Volume 4), which states that the ODYN approach is more conservative than the historical licensing philosophy for ATWS. Historically, for ATWS applications, prior regulatory approval has been granted for best-estimate code application based on the low probability of the event, conservatisms in key inputs and the acceptance criteria. [[

]]

The RAI response states that in overall, there is no additional specific treatment of uncertainties as ODYN was demonstrated to be conservative compared to test data and TRACG, and key inputs are set at conservative values.

The NRC staff accepts that the cited uncertainties treatment of valve tolerance and the specific key inputs are conservative, although nominal inputs are used in most parameters. This is acceptable for ATWS.

RAI 11-2: ODYN and TRACG Boron- Mixing Correlation

Provide the exact numerical values of the boron-mixing correlation used by TRACG and ODYN for ATWS calculations and their basis.

See Reference 31 RAI I-3.2 for RAI response.

<u>Evaluation RAI 11-2:</u>
TRACG uses the new mixing correlation developed at UC Santa Barbara in 1995. The values for the UC Santa Barbara correlation used in TRACG were confirmed to the NRC staff by Dr. Theofanous. ODYN uses the more conservative values from the old Vallecitos boron-mixing tests. The correlations used by the two codes are inconsistent, making performance comparison between the codes almost meaningless. However the correlation used by ODYN is the more conservative of the two as it bounds the results of the 1/6 scale tests at Vallecitos and the full-scale tests at Santa Barbara. Therefore, from the point of view of boron mixing correlation ODYN results should be more conservative than the best-estimate TRACG results. The correlation values use for both TRACG and ODYN are shown in the following tables.

Table A-8 Boron mixing correlations for lower plenum injection used by TRACG and ODYN

[[			
			]]

Table A-9 Boron re-mixing correlation used by TRACG and ODYN

[[			
			]]

RAI 11-3: ODYN limitations

What are the remaining limitations of the ODYN ATWS calculations (e.g., ATWS/stability)? How will those limitations be addressed (e.g., use of TRACG for ATWS/stability)?

See Reference 31 RAI I-3.5 for RAI response.

Evaluation of RAI 11-3:
The NRC staff agrees with GEH's evaluation of the limitations of the ODYN code for documentation purpose. ODYN calculations are limited by the following limitations:

1. The downcomer level must remain above the jet pump suction and no prolonged level in the active channel is allowed;
2. The duration of the simulation after the upper plenum subcools should be limited.
3. The mass in the separators should not remain zero and, therefore, the code is restricted to applications where the water level remains at or above the top of active fuel plus 5 feet;
4. The code is not presently qualified to perform stability calculations;
5. No lower plenum voiding is allowed.

Items 1 and 3 limit the ODYN level control strategy at TAF +5'. Item 5 prevents the simulation of depressurization with ODYN code.

The NRC staff agrees with GEH's evaluation that ODYN cannot perform the ATWS stability calculation. The ATWS stability calculation is performed with TRACG code as specified in the NRC approved NEDO-32047-A, "ATWS Rule Issues Relative to BWR Core Thermal-Hydraulic Stability."

RAI 11-4: Containment Performance During Isolation ATWS at MELLLA+ Conditions

Provide a comparison of ODYN results of isolation ATWS simulations at MELLLA+ and original licensed thermal power (OLTP).

See Reference 31 RAI I-6.1 for RAI response.

Evaluation of RAI 11-4:
Table A-10 below presents the information requested. The ODYN calculations indicate that containment over-pressure during ATWS is affected significantly by MELLLA+ operation. During the early part of the isolation transient, the peak vessel pressure increases by as much as 150 psi for some plants, because of limitations on maximum SRV flow. As shown in ATWS RAI 8 response, for a BWR4, the peak pressure reaches a value larger than the allowed 1500 psi assuming previously allowed one SRVOOS. The solution proposed in this case is to eliminate the flexibility to operate with one SRV out of service. If all SRVs are assumed operational for the specific BWR/4s analyzed, the calculated peak vessel pressure is less than the 1500 psi limit.

We conclude that MELLLA+ operation has a significant detrimental effect on the peak vessel pressure following an isolation ATWS. Because the margin to allowed peak pressure is so small, some plants may not be able to operate at MELLLA+ conditions. Under these circumstances, the licensees tend to re-evaluate their calculation assumptions and perform a new calculation with these new assumptions that satisfies the limits. The NRC staff must review these changes in assumptions to guarantee that they are within their allowed technical specifications. For example, it is acceptable to change Tech Specs to require all SRVs to be operable and then take credit for it. It is not acceptable to perform a probabilistic analysis and conclude that SRVs are available most of the time; therefore we can assume they are operable even though Tech Specs allows the plant to operate with a SRV out of service.

Limitation:

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

Table A-10 OLTP and MELLLA+ ATWS Results Comparisons

[[		
		]]

RAI 11-5: Actuation Sequence

For the above cases, provide the sequence of events (system and equipment actuation and operator actions for the mitigated cases) and the corresponding times. For example, for the MSIVC mitigated case, tabulate when the high pressure ATWS setpoint is reached, main steam isolation valve (MSIV) closes, ATWS-RPT occurs, peak vessel pressure is reached, feedwater(FW) reduction is initiated, boron injection initiation temperature (BIIT) is reached, SLC pumps starts, and water level increases.

See Reference 31 RAI I-6.2 for RAI response.

<u>Evaluation of RAI 11-5:</u>

The table below shows the system and component actuation setpoints for ATWS initiated from the MELLLA and MELLLA+ minimum flow statepoints. Note that the MELLLA and MELLLA+ minimum flow statepoints correspond to the approximately 120% and 140% rodlines as oppose to rated OLTP which corresponds to the 100% rodline.

Table A-11 BWR/4 (standpipe injection)

Item	Response	OLTP Time (sec)	M+ Event Time (sec)
1	[[		
2			
3			
4			
5			
6			
7			
8			
9			
10			
11			
12			
13			
14			
15			]]

Table A-12 BWR/6 (HPCS boron spray)

Item	Response	OLTP Time (sec)	M+ Event Time (sec)
1	[[		
2			
3			
4			
5			
6			
7			
8			
9			
10			

11			
12			
13			
14			]]

Note:
(1) For upper plenum boron injection plants, the water level stays at TAF or TAF+5' during the ATWS event. The operators do not need to raise water to promote boron mixing because the boron stratification is not an issue.

There appears to be inconsistencies in hot shutdown times for MELLLA and MELLLA+ operation calculations in which the reasoning is not clear. For the BWR/4, the MELLLA+ hot shutdown is achieved earlier ([[ ]]). The peak suppression pool occurs later ([[ ]]). It is not clear whether the hot shutdown is reached earlier because of higher Boron-10 concentration. Similarly, it is not clear why the peak suppression pool temperature occurs earlier for the BWR/6 ([[ ]]).

[[ ]] The RAI response (MFN 05-081) states [[ ]] The RAI response justifies the differences in the hot shutdowns times, [[ ]] the use of this criteria can cause variations in the reported time to hot shutdown.

The key inputs and assumptions used in the ATWS analyses are flagged as audit items. In plant-specific application review, the NRC staff is recommended to audit the assumptions and key inputs used in the ATWS analysis.

RAI 11-6: ATWS Response at OLTP, 5% Stretch, EPU and MELLLA+ Comparison

For all BWRs, tabulate the ATWS results (e.g., peak pressure, suppression pool temperature) before the 5 percent power stretch (if available), after the 5 percent power stretch (if applicable), and after EPU and EPU/MELLLA+. Include in the table the results from the initial GEH generic ATWS analyses. Since the initial plant licensing, many BWRs have adopted range-of operating condition changes that affect their ATWS response. These changes include increases in the fuel cycle length (cycle extension from 18 months to 24 months), power (from 5 percent to 20 percent uprates above the original licensed thermal power), and licensed operating domain (LLLL, ELLLA, MELLLA, maximum core flow). The objective of this table is to assess how the previous changes in the operating conditions affected BWR plants' ATWS margins. This would also serve as a means to evaluate the capability of BWRs to meet the vessel and containment response with the additional EPU/MELLLA+ changes. The NRC staff acknowledges that GEH may not have access to the plant-specific ATWS analysis-of-record for plants with other reload vendors.

See Reference 31 RAI I-6.3 for RAI response.

Evaluation of RAI 11-6:
Note that this RAI pre-dates the agreed upon resolution that TRACG sensitivity analysis will be performed.

The data provided indicate that peak vessel pressure is significantly affected by EPU/MELLLA+ operation (See Section I.6.1above). The suppression pool temperature calculated by the ODYN procedure does not appear to be affected as significantly. This is most likely caused by the extremely conservative nature of the ODYN calculation. However, the fact that the HCTL limit is reached and emergency depressurization will be required, makes it hard to compare suppression pool temperature calculations without depressurization.

Tables A-13 and A-14 present the peak vessel pressure and suppression pool data. Due to the historical ATWS code transition from REDY to ODYN, the results are not based on consistent methodology.

Table A-13 Vessel Pressure (psig) Database

[[				

]]

Table A-14 Suppression Pool Temperature (°F) Database

[[				
				]]

Note that Table A-14 shows that the suppression pool temperature reaches HCTL, even for operation at the MELLLA operating domain.

NRC RAI 12: IMPACT OF DEPRESSURIZATION ON CONTAINMENT AND CORE INTEGRITY

When following the EPGs, operators are required to depressurize the reactor if the HCTL is reached during the transient. The approved ODYN licensing methodology does not reflect this operator action (the suppression pool continues to heat up after HCTL is reached and the depressurization is ignored). Even though the ODYN licensing methodology may be conservative, ODYN results cannot determine whether the reactor fuel reaches PCT limits that may affect long-term coolability. Thus, TRACG calculations are required to evaluate the impact on fuel PCT limits of depressurization.

RAI 12-1: ODYN/TRACG Comparison

Provide the results of a set of TRACG calculations to evaluate the effect of the ODYN modeling limitations. Compare the TRACG results to the ODYN licensing calculation, including the PCTs. At a minimum, provide TRACG calculations based on limiting conditions that follow the EPGs (i.e., depressurization if HCTL is reached) at the three water level setpoints: TAF+5, TAF, and MSCWL and compare to the ODYN licensing methodology results.

See References 20 and 31 RAI I-3.3 for RAI response.

<u>Evaluation of RAI 12-1:</u>

TRACG sensitivity calculations with water level strategy at TAF+5, TAF, and TAF-2 were provided. These TRACG calculations followed the Emergency Operating Guidelines and core emergency depressurization was required at approximately 600 seconds into the transient, well before the Hot Shutdown Boron Weight (HSBW) was injected. Figure A-2 shows the integrated SRV flow for the three TRACG calculations and one ODYN calculation. The ODYN integrated SRV flow bounds the TRACG results for all three strategies. Note: these are TRACG calculations where the SRVs are assumed to remain open for the whole transient (after the 600 sec emergency depressurization is initiated). Other TRACG calculations were performed later where the SRVs are re-closed once 50 psig is reached per EOPs. Note that re-closing the SRVs results in a pressure perturbation that could induces re-criticality and the reactor power increases. Plant-specific application will provide the TRACG sensitivity analyses. The specific data provided in Figure A-2 indicates ODYN SRV flow is conservative in the latter phase of the event. Staff confirmatory analysis based on ODYN and TRACG SRV data used in CONTAIN do not show high level of differences in the ultimate suppression pool temperatures (See Figure 9-8 of the SE).

[[

]]

Figure A-2 Comparison of integrated SRV flow calculated by ODYN and TRACG at several water level strategies with emergency depressurization

[[

]]

Figure A-3 Comparison of peak pressure calculated by ODYN and TRACG at several water level strategies

Figure A-3 shows a comparison of the peak pressure calculated by TRACG and ODYN. ODYN has already been licensed for peak pressure calculations (by NEDE-24154-P-A, Licensing Topical Report, Qualification of the One-Dimensional Core Transient Model For Boiling Water Reactors Volume 3, Application of One-Dimensional Transient Model to Licensing Basis Transients August 1986). The NRC staff agrees with the conclusion that ODYN peak pressure calculations are conservative with respect to TRACG.

The ODYN PCT results were not presented by GEH as response to this RAI. Figure A-4 shows the PCT values calculated by TRACG for an isolation ATWS following the EOP procedures. This figure shows two significant temperature excursions: the initial excursions (~10 sec) which is related to the initial over pressurization transient, and the second excursion (~1000 sec) which is caused by a core dryout condition during the emergency depressurization. Note that the core dryout condition is more severe for the low water level control strategies (TAF-2) because the initial core coolant inventory is smaller when the emergency depressurization initiates.

[[

]]

Figure A-4 Peak fuel clad temperature calculated by TRACG for isolation ATWS. During emergency de-pressurization, the core uncovers and clad over-heating occurs, but clad temperature criteria are satisfied

The NRC staff disagrees with the generic disposition of the PCT issue. The plant-specific ODYN calculations cannot predict the depressurization dryout, but it can estimate the initial PCT excursion on a plant-specific basis. The resolution of this issue is documented Revision 2 of LTR.

RAI 12-2: Key Parameters during Depressurization Scenario

Based on the data provided above, demonstrate whether the approved ODYN ATWS methodology is conservative relative to TRACG analyses following the emergency operating procedures (EOPs). Compare the results of the ODYN and TRACG (at different water levels) in terms of meeting the ATWS acceptance criteria. Demonstrate that: (1) the TRACG sensitivity analyses and results are bounding or conservative for all the BWR fleet for EPU/MELLLA+ operating conditions, or (2) that the plant-specific ODYN analyses based on the TAF+5 water level strategy would bound the TRACG sensitivity analyses for all of the BWR fleet, or (3) propose a margin criteria for the ATWS acceptance criteria such that a TRACG analyses following the EOP would be performed for the plant-specific application if the margin criteria is not met. Provide at least the following parameters for the three water level strategies, core and vessel void fractions, fuel temperature profiles and time evolution, boron concentrations at several elevations in the lower plenum, recirculation flow, pressure, power levels, bypass voiding with in the vessel (lower and upper levels) and core wide reactivity.

See References 20 and 31 RAI I-3.4 for RAI response.

Evaluation of RAI 12-2:
The NRC staff agrees that the proposed plant-specific ODYN ATWS calculations bound the best-estimate TRACG results for:

1. Peak pressure calculations, and
2. PCT for the first temperature peak induced by the overpressure transient.

However, the NRC staff concludes that plant-specific ODYN ATWS calculations do not bound the best-estimate TRACG results for PCT temperature for the second temperature peak caused by core dryout during depressurization.

In addition, the ODYN ATWS calculations have both conservative and non-conservative assumptions, and bear little resemblance to the real ATWS transient. Comparison with sample TRACG best estimate ATWS calculations appears to indicate that the conservative assumptions compensate the non-conservative assumption and the overall ODYN result is likely to be conservative (i.e., yield a larger suppression pool temperature.) However, the NRC staff finds it hard to justify the use of the low-fidelity ODYN calculations to guarantee that the ATWS criteria are satisfied on a plant-specific basis. Therefore, the NRC staff does not concur with the proposed ODYN-based methodology for plant-specific ATWS analysis and recommends that best-estimate TRACG calculations be performed to confirm peak pressure.

At the request of the NRC staff, GE re-ran TRACG allowing the SRVs to reclose once 50 psig is reached per EOPs. Re-closing the SRVs results in a pressure perturbation that induces re-criticality and the reactor power increases. The results of these new calculations are shown as Figures 9-2 and 9-3 of the SE. The re-criticality periods are apparent in Figure 9-2 of the SE. They appear to be random in nature, amplitude and duration. Most have relatively low power levels (of the order of 20 to 30%), but some power spikes with power >100% are observed. Figure 9-3 of the SE shows that the reactor pressure during re-criticality periods is as high as 2 Mpa (300 psi), and it has some random characteristics. Figures 9-7 through 9-10 in this MELLLA+ SER show the integrated SRV flow and PCT, indicating that the re-criticality periods around 1500 seconds have a small effect on the overall suppression pool heat load or PCT. Note the large PCT transient in Fig 10 is during the de-pressurization stage and is caused by core dryout. The depressurization PCT response is expected to be more severe for the TRACG TAF-2 case, which is not currently available.

The ODYN calculation that was proposed in the original LTR uses a number of non-best-estimate assumptions that are mostly conservative. However, not all of these assumptions are conservative or even representative of real plant operation. For example, the ODYN calculation does not follow the EOP requirement to depressurize the reactor if the Heat Capacity Temperature Limit (HCTL) is reached - this is a non-conservative assumption. Nevertheless, we agree that the long-term integrated SRV flow calculated by the conservative ODYN procedure bounds the results of all the TRACG analysis performed. Since the ODYN calculation is plant specific, the results provide some value by having the actual plant parameters for suppression pool volume, SRV capacity, RHR performance.

However, the NRC staff concludes that ODYN does not model or estimate the physical phenomena that occurs during the depressurization phase. The reactor does not remain in a hot shutdown condition after the emergency de-pressurization. As can be seen from the total core reactivity plot, the reactor becomes critical and continues to be close to critical conditions after the hot shutdown boron weight is injected. The best-estimate TRACG calculations indicate some random behavior and large sensitivity to plant-specific assumptions, such as the specific ATWS water level control strategy (e.g., TAF+5' versus TAF-2') specified in the plant

EOPs. When re-criticality occurs, occasionally large power spikes (>100% nominal) are observed in some of the available runs (TAF+5 and TAF). In all cases, the reactor re-pressurizes to an intermediate pressure and remains at significant power for minutes and several cycles of de- and re-pressurization are observed for several minutes. The impact of the recriticality on the PCT and the long term cooling can only be determined by performing plant-specific calculations. For this reason, a limitation has been placed to perform best-estimate TRACG calculation on plant-specific basis. Section 9.3.1.3 of this SE provides additional discussion and an associated limitation.

RAI 12-3: Stages and Timing of the depressurization

Describe the stages and timing of the depressurization event that was modeled. Is boron mixing enhanced by this event using TRACG as opposed to the ODYN licensing methodology?

See Reference 31 RAI I-5.2 for RAI response.

Evaluation of RAI 12-3:
The data was provided. Boron mixing is enhanced by the depressurization, but the relative enhancement depends on the particular scenario. Specifically, in this simulations, the boron mixing was enhanced for the low water level scenarios (TAF-2 and TAF), but not for the high water level (TAF+5).

RAI 12-4: Core Void Fraction at Decay Heat Levels

Provide a series of steady-state sensitivity analyses to demonstrate that the core will remain subcritical following depressurization. Provide the core void fraction at decay heat levels and approximately 100 psi pressure for a range of core flows (e.g., 5 percent to 15 percent core flow) that could be possible depending on the water level control strategy.

See Reference 31 RAI I-5.3 for RAI response.

Evaluation of RAI 12-4:
The data was provided. ISCOR steady state calculations indicate that the average core void fraction at decay heat power level, 5% core flow, and 100 psi is approximately [[ ]]

which is sufficient to maintain the reactor subcritical if sufficient boron has already been diluted in the core. However, this void fraction can be reduced significantly if the recirculation core flow is increased, so re-criticality is not impossible. Indeed, TRACG and TRACE calculations both show the possibility of re-criticality occurring for most conditions.

NRC RAI 13: EMERGENCY AND ABNORMAL OPERATING PROCEDURES

RAI 13-1: Emergency and Abnormal Operating Procedures Affected by MELLLA+

Provide some specific examples where the EOPs would be affected by MELLLA+ operation. For example, a cursory review of the EPG/severe accident guidelines (SAGs) are examples of areas that need further evaluation and update for determining limiting values. Other variables not mentioned here may be affected.

1. Maximum Pressure for Heat Capacity Temperature Limit Plot (Section 17.5). Section 17.5 defines the procedure for calculation of the HCTL. In the example plots (Figures B- 17-5 and B-17-6), a maximum pressure of 1100 psig is used. However, TRACG calculations show that the pressure during MSIV ATWS is consistently above 1100 psig. Should the

EPG/SAGs be modified for EPU/MELLLA+ operation to require calculation of the HCTL at the expected higher pressures?

2. Hot Shutdown Boron Weight (HSBW) (Section 17.6). The first assumption is that the reactor is operating on the maximum extended operating domain. Clearly this assumption should be changed to the corner of the MELLLA+ domain. Assumption #6 specifies an operating pressure of 1100 psia. However, TRACG calculations show that during ATWS from EPU/MELLLA+ the expected pressures are significantly higher than 1100 psia.
3. Boron Injection Initiation Temperature. The BIIT is defined as the suppression pool temperature that will allow for injection of the HSBW without reaching the suppression pool HCTL. Should the BIIT curve be modified under MELLLA+ operation?
4. Minimum Number of Safety Relief Valves (SRVs) Required for Decay Heat Removal (Section 17.21). With EPU/MELLLA+, the expected decay heat levels should be higher. Will the minimum number of SRVs change? Will this number affect any other variables?
5. Minimum Number of SRVs Required for Emergency Depressurization (Section 17.22). With EPU/MELLLA+, the expected ATWS power levels should be higher. Will the minimum number of SRVs change? Will this number affect any other variables?
6. Minimum Steam Cooling Pressure (Section 17.23). With EPU/MELLLA+, the expected ATWS power levels should be higher. Will the minimum steam cooling pressure be higher? If the pressure is higher, will this affect any other variables?
7. Minimum Steam Cooling Reactor Pressure Vessel (RPV) Water Level (Section 17.24). With EPU/MELLLA+, the expected ATWS power levels should be higher. Will the minimum steam cooling RPV water level change? If the level does change, how does it affect any other variables?
8. Minimum Zero-Injection RPV Water Level (Section 17.25). With EPU/MELLLA+, the expected ATWS power levels should be higher. Will the minimum zero-injection RPV water level change? If the level does change, how does this affect any other variables?

See Reference 31 RAI I-7.1 for RAI response.

<u>Evaluation of RAI 13-1:</u>
The NRC staff agrees with GEH's evaluation and includes a limitation requiring the review of the EPG/SAG parameters in a plant specific basis.

<u>Limitation</u>:

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

RAI 13-2: EOP Guidance

Since most of these parameters are likely to be affected by MELLLA+ operation in all plants, provide the justification why the LTR does not provide generic guidance on these parameters.

See Reference 31 RAI I-7.2 for RAI response.

<u>Evaluation of RAI 13-2:</u>
GEH states that The LTR does not provide generic guidance because the BWROG is the owning body for the EPG/SAG, including the technical bases descriptions, and they have already completed a generic evaluation of the EOP curves and limits that are potentially

affected by changes to reactor power and operating domain. This is fully sufficient to ensure that plant EOPs are updated appropriately for MELLLA+ implementation.

The NRC staff agrees with GEH's position and, therefore, includes a limitation requiring the review of the EPG/SAG parameters in a plant specific basis

NRC RAI 14: ATWS/STABILITY ANALYSES

A major concern for the nonisolation turbine trip ATWS is the presence and impact of unstable large power oscillations, which occur when the flow is reduced and the feedwater temperature cools down as a result of the turbine trip. To manage the consequences of these large power oscillations, the EPGs prescribe a number of mitigation actions intended primarily to suppress these oscillations, including reduction of water level below the feedwater sparger and early boron injection. MELLLA+ operation increases the operating control rod line and increases the likelihood and the resulting amplitude of large power oscillations during ATWS events. The relevant question is whether the EPG mitigation actions are still effective under MELLLA+ conditions.

RAI 14-1: Non-Isolation ATWS with Mitigation

Provide the results of a TRACG calculation for a non-isolation ATWS with the prescribed mitigation actions. Compare to the TRACG results without mitigation actions. Provide the fraction of the core that reaches PCT limits during the non-isolation ATWS with and without mitigation actions.

See Reference 31 RAI I-4.1 for RAI response.

<u>Evaluation of RAI 14-1:</u>
The requested calculation was provided by GEH. Since the transient analyzed is a non-isolation ATWS, the condenser is available and the suppression pool temperature does not change. The main concern here is whether the mitigation actions prescribed by the Emergency Operating Guidelines are still effective to mitigate the instability when the reactor initial condition is within the MELLLA+ domain.

[[

]]

Figure A-5 Average core power and flow following a non-isolation turbine trip with bypass ATWS. (No EOPs stability mitigation actions followed)

Figure A-5 shows the average core power and flow following a non-isolation turbine trip with bypass ATWS where the EOP-mandated mitigation actions were not followed. This transient was determined to be the bounding case for oscillation amplitude because the turbine trip cuts off the supply of steam to the feedwater heaters, causing the feedwater temperature to drop to condensate storage tank temperature. This causes a large increase in core inlet subcooling, which results in a very significant increase in power level. At the resulting power levels, the instability grows to a large amplitude limit cycle.

This scenario was analyzed in 1992 and reported in NEDO-32164, and the consequences were found unacceptable by both the industry and the NRC staff. As a result the Emergency Procedure Guidelines (EPGs) were updated to include stability mitigation actions. These mitigation actions include: early boron injection and early water level reduction to below the feedwater spargers to preheat the incoming cold feedwater with vessel steam.

[[

]]

Figure A-6 Average core power and flow following a non-isolation turbine trip with bypass ATWS. (Performed following EOPs water level is lowered below the FW spargers)

The key issue is whether the mitigation actions studied in NEDO-32164 and implemented in EOPs are still effective under MELLLA+ operating domain. To this end, GEH performed a TRACG calculation following the EOP mitigation actions. The results are shown in Figure A-6. For this transient the water level is lowered to TAF at time 120 seconds, to simulate a typical operator action delay. As seen in this figure, the water level reduction and early boron injection mitigate the unstable oscillations in approximately 80 seconds after operator action is initiated.

The EOPs require that the water level be lowered to at least 2 ft below the sparger. When the cold feedwater enters the downcomer, a small fraction wets the vessel surfaces and runs down along them. The remaining water is dispersed in the downcomer steam environment and exchanges enthalpy rapidly with the steam phase. Thus, lowering the water level has two effects: (1) it preheats the feedwater before it enters the core, thus reducing the effective core reactivity and thermal power generation. (2) It condenses dome steam, thus reducing the steam line flow and the heat load to the suppression pool. Lowering the water level is the most effective prompt-effect action that can be taken during ATWS.

Figures A-7 and A-8 present the limiting bundle powers and PCT. As can be seen from these plots limited fuel damage is expected. Since for the EPU/MELLLA+ operating strategies, number of maximum powered bundles increase, the % of core experiencing fuel damage is expected to increase. Figures A-9 and A-10 provide the core and vessel parameters through the course of the scenario, showing the changes in the inlet subcooling, vessel water level and the FW temperature and flow. As modeled, the FW temperature decreases as the preheating is lost and the FW temperature equilibrates to the condenser temperature. Figure A-11 shows the vessel pressure.

For non-isolation ATWS with instability, the NRC staff concluded that:

1. Operation in MELLLA+ is detrimental to ATWS/Stability, because it increases the effective rod line following the recirculation pump trip; thus increasing the ATWS power level, the probability on instabilities during ATWS, and the probability that the unstable oscillations will grow to very large amplitudes
2. However, the ATWS/Stability mitigation actions prescribed in the EPGs are still effective even when operating the reactor in the MELLLA+ domain. Thus, large amplitude unstable oscillations will be mitigated in relatively short time by operator actions prescribed in the EPGs.

[[

]]

Figure A-7 Limiting Bundle Power (Based on EOP Actions)

[[

]]

Figure A-8 Limiting Bundle PCTs (Based on EOP Actions)

[[

]]

Figure A-9 Core And Vessel Parameters (Based on EOP Actions)

[[

]]

Figure A-10 FW Temperature and Core Inlet Subcooling (Based on EOP Actions)

[[

]]

Figure A-11 Core Pressure (Based on EOP Actions)

RAI 14-2: Power and PCT for Implicit and Explicit Numeric

Provide the results of a TRACG calculation for a full-isolation ATWS with depressurization using the TRACG stability numerics.

See Reference 31 RAI I-4.2 for RAI response.

Evaluation of RAI 14-2:
The core power and PCT responses were provided for both, the implicit numeric scheme and the explicit (stability) numerics. Figures A-11 and A-12 show that the differences are not significant. Note that the analysis does not include depressurization.

[[

]]

Figure A-12 Reactor Power

[[

]]

Figure A-13 PCT

RAI 14-3: Effectiveness of the Mitigation Actions

Are the mitigation actions prescribed by the EPGs effective to manage ATWS/Stability concerns under MELLLA+ operating conditions?

See Reference 31 RAI I-4.3 for RAI response.

Evaluation RAI 14-3:

Based on the results of the above analysis, the NRC staff agrees with the conclusion that the operator actions prescribed in the emergency procedure guidelines (EPGs) are effective in managing ATWS instability concerns under MELLLA+ operating conditions. It is the responsibility of the fuel vendors and licensees to ensure that this conclusion remains applicable and valid for future changes of fuels design and/or operating strategy. The basis for the current analysis can be found in the RAI responses and the topical report.

RAI 14-4: Documentation

Table 9-5 of the MLTR lists the fuel response for the set of ATWS instability analyses. Figures 9-5 to 9-11 of the MLTR show the fuel response for the high-powered bundles. For clarity, add sub-titles or footnotes to the figures that identify the statepoints and the initial power to flow conditions. Otherwise, label Table 9-5 and the corresponding figures by case numbers. Expand Table 9-5 to include event type (turbine trip or MSIVC) and the mitigated cases. Footnote the mitigation strategy used.

See Reference 31 RAI II-1.1 for RAI response.

Evaluation of RAI 14-4:
The revised LTR would contain the requested information.

RAI 14-5: Documentation of the Analyzed Conditions

Footnote 2 to Table 9-5 of the MLTR states: [[

]] Please, explain this statement. [[

]]

See Reference 31 RAI II-1.2 for RAI response.

Evaluation RAI 14-5:
GEH's evaluation of ATWS/Stability was performed for [[
]] which appears to bound all the reactors in the fleet. The NRC staff agrees that this evaluation is sufficiently conservative for all other operating conditions with a smaller initial power density. Other initial conditions (e.g. 120% OLTP and 100%, or lower power-density reactors) are likely to exhibit a less severe stability response during ATWS events.

Limitation:

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

RAI 14-6: Percent Fuel Failure Rate

Since for EPU/MELLLA+ core design, the number of high-powered bundles will increase, provide an estimate of the percent of the core that may experience PCT greater than 2200° F for the unmitigated cases. Compare this with the conclusions reached from the original ATWS instability evaluations in Reference 14 of the MLTR.

See Reference 31 RAI II-1.3 for RAI response.

Evaluation RAI 14-6:
For the present MELLLA+ evaluation, [[ ]] For comparison, 12 percent of the bundles exceeded the limit in the original ATWS instability evaluations in Reference 14. The NRC staff notes that this difference is not statistically significant, because the fuel failures occur during a single unstable pulse (i.e., a power excursion of < 1 sec). During large-amplitude instability events, the pulse amplitudes are essentially random in nature, and the percentage of fuel that exceeds the limit is also random. Considering that the analyses results are based on different analysis conditions and model cores loaded with different fuel designs, the usefulness of the comparisons are limited.

RAI 14-7: Conservatism of the [[ ]]

Considering the variation that exists through the BWR fleet, explain why the [[ ]] is considered to be reasonably bounding.

See Reference 31 RAI II-1.5 for RAI response.

Evaluation of RAI 14-7:
GEH states that a power to flow ratio of [[ ]] is higher than any operating BWRs. The NRC staff has placed a limitation of [[ ]] for the applicability of these analyses. This limitation must be verified for all plant-specific applications.

RAI 14-8: Limiting Plant Configuration

Discuss the scoping criteria, if any, used to select the combination of limiting BWR plant physical configuration characteristics and operating parameters. Explain why [[ ]] was selected for performing the ATWS instability analyses. Include in the discussion the bases for selecting [[ ]] in terms of bypass, FW capacity and type, SRV capacity, and fuel support orifice size. Explain how the limiting power distribution (radial and axial), core loading pattern and core exposures, and the initial minimum critical power ratio were selected in order to analyze the bounding ATWS instability cases for the MELLLA+ operation.

See Reference 31 RAI II-1.2 for RAI response.

Evaluation of RAI 14-8:
ATWS instability events are evaluated to assure that the core coolable geometry criterion is met. These evaluations indicate that ATWS/Stability events from EPU/MELLLA+ conditions clearly violate the criterion if mitigation actions are not employed. Note that this conclusion is not limited to EPU/MELLLA+. ATWS/Stability events without mitigation actions also violate fuel limits when operating at OLTP conditions. The key question is whether the mitigation actions are still effective under EPU/MELLLA+ conditions.

The NRC staff agrees that the use of the [[ ]] TRACG deck is reasonably conservative to perform the above evaluation. This deck includes [[ ]] All these characteristics tend to increase the amplitude of the resulting unstable power oscillations.

RAI 14-9: Conservatism of the Use of GE14 Fuel

[[

]] Compare the instability response of the different GE fuel product line.

See Reference 31 RAI II-1.7 for RAI response.

Evaluation of RAI 14-9:
GEH states that the intended fuel for MELLLA+ application is GE14. To address transition cores, [[

]] The NRC staff concurs with GEH's evaluation that, with the mitigation strategies recommended by the EOPs, the core coolable geometry criterion would not be compromised following an ATWS instability event assuming any current GE fuel designs up to GE14.

Limitation:

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

RAI 14-10: Core and Fuel Design Dependency of ATWS Instability

Provide the bases and technical justifications that demonstrate [[ ]] response to an ATWS instability event will be bounding in comparison to the response for cores loaded with non-GE fuel, new GE fuel, or mixed cores. Alternatively, provide the licensing limitation that would be necessary for operation along the MELLLA+ boundary, unless specific ATWS instability analyses are provided for cores loaded with non-GE fuel or new GE fuel. Explain what analyses would be required if a plant licensed for operation along the MELLLA+ rodline, was loaded with non-GE fuel (e.g., SVEA 96 or ATRIUM 10) or new Global Nuclear Fuel (GNF) fuel.

See Reference 31 RAI II-1.8 for RAI response.

Evaluation of RAI 14-10:
The NRC staff agrees with GEH's evaluation that the fuel response to ATWS instability is dependant on the assumed fuel design, and that, for plants planning to implement MELLLA+ with a different vendor's fuel or future fuel design beyond GE14, additional justification is required to assure that the core coolable geometry criterion is met with the non-GE fuel design or future GE fuel design.

GEH has committed to perform a confirmatory ATWS instability analysis to justify a different vendor's fuel design or future fuel design beyond GE14. This analysis will be performed with TRACG (or equivalent analytical model) and will simulate the limiting TTWB event resulting in regional oscillation mode. Evaluation of RAI 14-9 provides additional discussion and an associated limitation.

RAI 14-11: Modeling Assumptions - Debris Filters

Were the fuel debris filters modeled in the ATWS analyses? If the fuel debris filters were not included in the analyses supporting MELLLA+ ATWS, explain the reason why the debris filters and the corresponding pressure drops were not included in the analyses. Justify why the results are acceptable. Alternatively, please provide the results of sensitivity analyses that demonstrate the impact of the debris filters on the plant's response to an ATWS. Similar effects should be described for transient analyses.

See Reference 31 RAI II-1.9 for RAI response.

Evaluation of RAI 14-11:
Additional pressure drop caused by the fuel debris filters was not included in the calculation. The NRC staff agrees that lowering the assumed core inlet friction enhances unstable power oscillations and, thus, is a conservative assumption with respect to stability. However, debris filters may reduce the natural circulation state flow after RPT, therefore making the reactor more unstable. Thus, their total effect must be considered.

Limitation:

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

NRC RAI 15: INCREASED PROBABILITY OF STARTUP INSTABILITIES

The WNP-2 (Columbia) instability event was caused primarily by an extremely skewed radial power distribution, which was achieved by withdrawing most of the hot-channel control rods early during the startup process. Following the instability event, GEH recommended that hot channel control rods not be withdrawn fully until after the pump up-shift maneuver, when the reactor is more susceptible to startup instabilities. In consideration that a MELLLA+ design core will have significantly more hot channels, two issues need to be addressed:

1. Are the radial power distributions likely to be more skewed during startup (as in the Columbia event) because there are so many hot channels that the operator will have to withdraw the control rods?
2. Will guidance be provided to utilities and operators that startup control rod patterns that have worked in the past may result in instabilities during normal control rod maneuvers?

See Reference 31 RAI II-1.10 for RAI response.

Evaluation of RAI 15:
GEH agrees that the EPU core designs result in a greater number of bundles near the maximum power and adds that the MELLLA+ core design does not, strictly speaking, need higher power peaking than MELLLA core designs. Therefore, operators of both EPU and MELLLA+ plants must take proper care not to increase the probability of startup instabilities

NRC RAI 16: SAFETY SYSTEMS ACTUATION LIMITS

RAI 16-1: NPSH of Safety Systems that Depend on Suppression Pool Water

What are the net positive suction head (NPSH) limits for safety systems that depend on suppression pool water (e.g. RHR, high pressure cooling injection (HPCI), etc)?

See Reference 31 RAI III-1 for RAI response.

Evaluation of RAI 16-1:

NPSH Design for Accidents

For accidents, GEH states that the NPSH limits for safety systems that take suction from the suppression pool during an accident are different for plants with different vintage and their licensing commitments. The plants can be grouped into two categories: Pre-Regulatory Guide 1.1 design (BWR/3 and early BWR/4 plants) and Regulatory Guide 1.1 design (late BWR/4 and BWR/5,6 plants).

Pre-Regulatory Guide 1.1 Design

[[

]]

Regulatory Guide 1.1 Design

For plants in the category, the NPSH design does not depend on containment overpressure credit. It assumes 0 psig containment pressure and the highest suppression pool temperature. For example, [[
]]

NPSH Limit, During Non-Accident Events

For accidents, NRC accepts the NPSH designs with limited or no containment overpressure credits. In the RAI response, GEH states that for non-accident events such as ATWS, NRC does not limit the credit for containment overpressure. Staff does not concur with this statement in that the NRC staff has, thus far, taken explicitly different position with regard to NPSH credit for non-accident events such as ATWS.

For non-accident events such as the ATWS, the NPSH limit depends on the availability of containment overpressure. The NRC staff finds that the containment atmosphere heats up and pressurization rates are different from LOCA and ATWS. The ATWS containment pressurization will occur at a slower rate. Therefore, when containment over-pressure credit is taken for ECCS equipment, the containment calculations must use modeling assumptions that include the slow containment pressurization rate. The LOCA containment conservative assumptions are likely not conservative for ATWS containment over-pressure credits. For example, The STEMP calculations discussed in RAI 16-3 assumes that the peak drywell pressure is equal to the peak wetwell (suppression pool) pressure. This assumption may not be conservative if the NPSH overpressure credit containment analysis is based on similar assumption.

<u>NPSH Limit for HPCI pumps</u>

The NPSH specification for the HPCI pumps requires a suppression pool temperature lower than 140°F, which is based on the HPCI pump lube oil system temperature limit. Since the HPCI operation temperature limit is 140°F, then it is also the NPSH limit for the HPCI pumps. This limit applies to both accident and non-accident events for the high pressure ECCS HPCI system. Other lower-volume equipment can operate with temperatures as high as 212°F.

These numbers are plant-dependent and may depend on containment over-pressurization credits. For example, with 5 psig containment credit, Plant D can operate the HPCI pumps with inlet water up to 170°F for short periods of time. For Plant D, the ODYN suppression pool temperature is 198 F, which occurs at 1770 seconds. The hot shutdown is reported to occur at 1408 seconds. Therefore, NPSH evaluation for all safety systems relied upon to provide makeup water for the event duration is necessary.

Limitation:

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

RAI 16-2: HPCI Maximum Back-Pressure

The pressure during ATWS events oscillates as high as 1200 psi for long periods (>20 minutes). Is HPCI capable of injecting sufficient volume with such high backpressure? Are any other safety systems affected by a 1200 psi backpressure?

See Reference 31 RAI II-2 for RAI response.

Evaluation of RAI 16-2
HPCI and RCIC are constant flow systems up to the capability of the turbine controls. The maximum operating pressure that is quoted in some documents refer to the maximum pressure below which HPCI and RCIC can deliver full flow. At higher pressures, HPCI and RCIC perform as constant speed centrifugal pumps; therefore, as pressure increases above design, pump flow rates may be expected to decrease accordingly, but the total injection flow rate would still be substantial. Nevertheless, this must be evaluated on plant-specific bases.

RAI 16-3: Containment Pressure

The STEMP results show containment pressurizations as high as 12 psig. Do such high containment pressures affect the actuation of any safety grade systems in the containment such as air-actuated valves?

See Reference 31 RAI II-3 for RAI response.

Evaluation of RAI 16-3
Section 10.3 of the MELLMLTR covers the assessment of environment on the qualification and function of safety systems. The plant-specific application will include confirmation that existing environmental envelopes for safety grade equipment remain valid with EPU and MELLLA+ conditions. The limiting pressures in the containment occur during DBALOCA, which bound the pressures resulting from the ATWS event. GEH provided generic evaluation comparing the DBA-LOCA and ATWS peak drywell and wetwell pressures. Table A-15 provides comparisons of the peak containment and suppression pool pressures.

Table A-15 DBA-LOCA and ATWS Containment and Suppression Pool Peak Pressure Comparisons

	Brunswick MELLLA+/EPU Mark I Containment	**Representative Mark II Containment (PU)**	**Representative Mark III Containment (MELLLA+/EPU)**
DBA-LOCA Peak Drywell Pressure (psig)	46.4	39.9	23.2
ATWS Peak Drywell Pressure* (psig)	12.7*	13.7*	7.2*
DBA-LOCA Peak Wetwell Pressure (psig)	31.1	27.9	7.0 **
ATWS Peak Wetwell (psig)	12.7*	13.7*	7.2*

*Note The ATWS analysis assumes that the peak drywell pressure is equal to the peak wetwell pressure calculated with the STEMP code.
** This is the peak pressure which occurs in the containment airspace above the HCU floor.

<u>Limitation</u>:

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

NRC RAI 17: PLANT E -SPECIFIC REQUEST FOR ADDITIONAL INFORMATION

Plant E is also included in EPU/MELLLA+ reference plants for the evaluating the ATWS response. This section of the RAI addresses specific assumptions used for these analyses.

RAI 17-1: Plant E Suppression Pool Limits

Justify the use of the 185°F ATWS suppression pool temperature limit for the EPU/MELLLA+ ATWS analysis. Specifically, justify why the suppression pool temperature limit is higher than the temperature limit required for depressurization.

See Reference 31 RAI V-1 for RAI response.

<u>Evaluation of RAI 17-1:</u>
GEH chose not to address this question in a generic basis, citing that it is best discussed on a plant-specific basis.

<u>Limitation</u>:

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

RAI 17-2: ATWS Transients Following the EOPs

The peak suppression pool temperature for EPU/MELLLA+ reported in NEDC-33057P is 171° F. While this number is below the reported 185°F limit, the reactor is still at full pressure. Thus, the reported 171° F is not the peak temperature, but the initial condition prior to depressurization. It would appear that following a depressurization (which is required by the EOP at this temperature), the suppression pool temperature would be greater than 185° F. Please provide the actual peak suppression pool temperature when the ATWS transient is followed to completion according to the EOPs.

See Reference 31 RAI V-2 for RAI response.

Evaluation of RAI 17-2:
Even though GEH chose not to answer this RAI, the ATWS transient following the EOPs was provided in the generic TRACG analysis responses. Note: these RAIs apply only to plants with standpipe boron injection. For plants that spray boron in the upper plenum, boron stagnation is not an issue, and the reactor achieves shutdown through borated water spray promptly. However, for BWR5 and 6 (HPCS plants) such as Plant E, the containment limits are significantly smaller and may be breached in the short times that it requires to achieve hot shutdown. Therefore, a plant specific calculation is also required for these plants.

RAI 17-3: EOPs

The effect of EPU/MELLLA+ on EPG/SAGs. Provide a critical review of the EPGs/SAGs to determine which variable definitions and calculations are affected by EPU/MELLLA+. The following sections provide some examples of areas that need further evaluation and update for determining limiting values. Other variables not mentioned here may be affected.

1. Maximum Pressure for Heat Capacity Temperature Limit Plot (Section 17.5). Section 17.5 defines the procedure for calculation of the HCTL. In the example plots (Figs. B-17-5 and B-17-6) a maximum pressure of 1100 psig is used. However, TRACG calculations show that the pressure during an MSIV ATWS is consistently above 1100 psig. Should the EPG/SAGs be modified for EPU/MELLLA+ operation to require calculation of the HCTL at the expected higher pressures?
2. Hot Shutdown Boron Weight (Section 17.6). The first assumption is that the reactor is operating on the maximum extended operating domain. Clearly, this assumption should be changed to the corner of the MELLLA+ domain. Assumption #6 specifies an operating pressure of 1100 psia. However, TRACG calculations show that during ATWS under EPU/MELLLA+ conditions the expected pressures are significantly higher than 1100 psia.
3. Minimum Number of SRVs Required for Decay Heat Removal (Section 17.21). With EPU/MELLLA+, the expected decay heat levels should be higher. Will the minimum number of SRVs change? If the minimum of SRVs does change, will this affect any other variables?
4. Minimum Number of SRVs Required for Emergency Depressurization (Section 17.22). With EPU/MELLLA+, the expected ATWS power levels should be higher. Will the minimum number of SRVs change? If the minimum number of SRVs does change, will this affect any other variables?
5. Minimum Steam Cooling Pressure (Section 17.23). With EPU/MELLLA+, the expected ATWS power levels should be higher. Will the minimum steam cooling pressure change? Will this pressure change affect any other variables?

6. Minimum Steam Cooling RPV Water Level (Section 17.24). With EPU/MELLLA+, the expected ATWS power levels should be higher. Will the minimum steam cooling RPV water level change? If the level does change, will this affect any other variables?
7. Minimum Zero-Injection RPV Water Level (Section 17.25). With EPU/MELLLA+, the expected ATWS power levels should be higher. Will the minimum zero-injection RPV water level change? If the water level changes, will this affect any other variables?

See Reference 31 RAI V-4 for RAI response.

Evaluation of 17-3:
GEH chose not to address this question on generic bases, citing that it is best discussed on a plant specific basis.