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MFN 04-027
March 10, 2004

U.S Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20852-2738

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

Subject: **Response to MELLLA Plus ATWS RAIs (TAC No. MB6157)**

By Reference 1, the NRC requested additional information (RAI) in order to support their review of the Licensing Topical Report (LTR) related to the NEDC-33006P, Revision 1, *Licensing Topical Report NEDC-33006P, Revision 1, "General Electric Boiling Water Reactor Maximum Extended Load Line Limit Analysis Plus."* The RAIs addressed by Reference 1 pertain to the review of Anticipated Transients Without Scram (ATWS) analyses. The response to each of these RAIs is enclosed.

A non-proprietary version of the requested information is provided in Enclosure 1. The responses with the proprietary information, as defined by 10CFR2.390, are provided in Enclosure 2. GE customarily maintains this information in confidence and withholds it from public disclosure.

The affidavit contained in Enclosure 3 identifies that the information contained in Enclosure 2 has been handled and classified as proprietary to GE. GE hereby requests that the information of Enclosure 2 be withheld from public disclosure in accordance with the provisions of 10 CFR 2.390 and 9.17.

If you have any questions, please contact, Mike Lalor at (408) 925-2443 or myself.

Sincerely,

James F Harrison for

George Stramback
Manager, Regulatory Services
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D065

Project No. 710

Reference:

1. MFN 04-008, Letter from Alan Wang (NRC) to James Klapproth (GE), January 30, 2004, *Request for Additional Information - Licensing Topical Report NEDC-33006P, Revision 1, "General Electric Boiling Water Reactor Maximum Extended Load Limit Analysis Plus," (TAC No. MB6157)*

Enclosures:

1. Response to NRC MELLLA+ ATWS RAIs – – Non-Proprietary Information
2. Response to NRC MELLLA+ ATWS RAIs – –Proprietary Information
3. Affidavit

cc: AB Wang (NRC)
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ENCLOSURE 3

MFN 04-027

AFFIDAVIT

General Electric Company

AFFIDAVIT

I, David J. Robare, state as follows:

(1) I am Technical Projects Manager, Technical Services, General Electric Company ("GE") and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in Enclosure 2 to GE letter MFN 04-027, George Stramback to NRC, *Response to MELLLA Plus ATWS RAIs (TAC No. MB6157)*, dated March 10, 2004. For text and text contained in tables, the GE proprietary information in Enclosure 2 is identified by a double underline inside double square brackets. Proprietary figures and large equation objects are identified with double square brackets before and after the object. In each case, the superscript notation{3} refers to Paragraph (3) of this affidavit, which provides the basis for the proprietary determination.

(3) In making this application for withholding of proprietary information of which it is the owner, GE relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), and 2.390(a)(4) for "trade secrets" (Exemption 4). The material for which exemption from disclosure is here sought also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by General Electric's competitors without license from General Electric constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals aspects of past, present, or future General Electric customer-funded development plans and programs, resulting in potential products to General Electric;

d. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (4)a., and (4)b, above.

(5) To address 10 CFR 2.390 (b) (4), the information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GE, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GE, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within GE is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his delegate), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GE are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information identified in paragraph (2), above, is classified as proprietary because it contains detailed results and conclusions from evaluations of the safety-significant changes necessary to demonstrate the regulatory acceptability for the expended power/flow range of MELLLA+ for a GE BWR, utilizing analytical models and methods, including computer codes, which GE has developed, obtained NRC approval of, and applied to perform evaluations of transient and accident events in the GE Boiling Water Reactor ("BWR"). The development and approval of these system, component, and thermal hydraulic models and computer codes was achieved at a significant cost to GE, on the order of several million dollars.

The development of the evaluation process along with the interpretation and application of the analytical results is derived from the extensive experience database that constitutes a major GE asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GE's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GE's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GE.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

GE's competitive advantage will be lost if its competitors are able to use the results of the GE experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GE would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GE of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information, and belief.

Executed on this 10 day of MARCH 2004.

David J. Robare
General Electric Company

ENCLOSURE 1

MFN 04-027

Response to NRC MELLLA+ ATWS RAIs

Redacted and Non-Proprietary Information

Table of Contents

Table of Contents (cont.)

Part I, Section 9.3.1, "Anticipated Transients Without Scram"

NRC RAI I-1.0, ATWS Events

In establishing the ATWS events that would be analyzed on plant-specific bases, the MELLLA+ licensing topical report (MLTR) states that the limiting ATWS event for the containment response depends on the [[

]] The following questions address the bases for these conclusions.

NRC RAI I-1.1 LOOP

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 the [[

]]

GE Response

The Loss of Offsite Power event cuts off the power supply to the motor-driven pumps and initiates a turbine-generator trip immediately. Consequently, it initiates the following system response:

[[

]]

NRC RAI I-1.2, Inadvertent Opening of Relief Valve (IORV)

The 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.

GE Response

[[

]] The operators would first initiate a scram, but this is assumed to fail. [[

]]

The reactor vessel and fuel integrity is not challenged during the IORV event. The main impact to the reactor system is the heatup of the suppression pool for the constant injection of steam from the reactor vessel. [[

]] The operator can initiate the controlled blowdown after hot shutdown is achieved. The amount of steam and energy from the blowdown process is similar for all the ATWS events because the energy remaining in the vessel is similar for all events at the time of hot shutdown. [[

]]

NRC RAI I-2.0, Determining the Peak Clad Temperature (PCT)
The MLTR states that [[

]] The following questions are related to the PCT.

NRC RAI I-2.1
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?

GE Response
[[

]]

The ATWS event response is more severe than an AOO event that results in a scram. The ATWS event is categorized as a special event and the acceptance criterion is not the SLMCPR criterion applied to the AOOs. From Section 9.3.1 of the M+LTR the acceptance criteria are as follows.

The ATWS evaluation will be performed using the methodology documented in Section 5.3.4 of ELTR1 and will meet the following criteria:

- Maintain reactor vessel integrity (i.e., peak vessel bottom pressure less than the ASME service level C limit of 1500 psig).
- Maintain containment integrity (i.e., maximum containment pressure and temperature lower than the design pressure and temperature of the containment structure).
- 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. Previous ATWS analyses have demonstrated significant margin to the acceptance criteria of 10CFR50.46.

NRC RAI I-2.2

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.

GE Response

The following table outlines all the PCT results from the ODYN/ISCOR/TASC methodology:

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

[[					

]]

NRC RAI I-2.3

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.

GE Response

[[

]] The operating conditions listed in RAI 2.20 response cover the range of ELLLA to MELLLA+ and 100% Original Licensed Thermal Power (OLTP) to 120% OLTP (EPU). The fuel types evaluated include GE9, GE13 and GE14. [[

]] PCT values during the ATWS events would meet the 2200°F limit and the oxidation would be insignificant.

NRC RAI I-2.4

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.

GE Response

[[

]] Therefore, ATWS analysis is performed at the minimum core flow state point.

NRC RAI I-3.0, Applicability of the ODYN Licensing Methodology to ATWS Analyses
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.

NRC RAI I-3.1
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.

GE Response
NEDC-24154P-A, Revision 1, February 2000, Qualification of the One-Dimensional Core Transient Model (ODYN) for Boiling Water Reactors (Supplement 1 -Volume 4), provides the description of the NRC approved ODYN ATWS methodology and assumptions that limit the application range of ODYN. Section 4.1 from the NRC Safety Evaluation for this Supplement (reproduced below) identifies the restrictions on ATWS applications.

> 4.1 Application Scope of ODYN
> GENE proposes to expand the scope of ODYN applications to include ATWS and non-pressurization transients. The LTR under review contains ODYN results to validate its application to ATWS and a discussion and example evaluation of the proposed method for calculating non-pressurization transient ΔCPR. GENE proposes to follow a conservative method for non-pressurization transient ΔCPR calculations discussed in section 5 of the report (ref. 1). Prior to the current modifications ODYN was fully capable of predicting non-pressurization transients and with the current modifications it is also capable of predicting ATWS conditions. During the course of generation of the LTR and the staff review, GENE identified the following restrictions on ODYN ATWS applications:
>
> a. The downcomer level must remain above the jet pump suction and no prolonged level in the active channel is allowed:
> b. The duration of the simulation after the upper plenum subcools should be limited.
> c. 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.
> d. The code is not presently qualified to perform stability calculations.
> e. No lower plenum voiding is allowed.

Section 5.4 of NEDC-24154P-A, (Supplement 1-Volume 4), Input Parameters/Event Simulation, provides a list of the key parameters for ATWS and non-pressurization transients, as an extension of Volume 3, NEDE-24154P-A, Section 3. The NRC Safety Evaluation for Volume 1, NEDO-24154-A, Volume 2, NEDC-24154-A, and Volume 3, NEDE-24154P-A, August 1986 provides a summary of code uncertainties in Section 6.

NRC RAI I-3.2

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

GE Response

The following two tables list the boron mixing and remixing coefficients used in the ATWS calculation. The ODYN coefficients are adjusted based on the comparison of its integrated SRV flow against that from the TRACG calculation for the same event. The TRACG coefficients are based on the University of California - Santa Barbara full scale boron mixing test data.

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			]]

NRC RAI I-3.3

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.

GE Response

The dome pressure and integrated SRV flow comparisons between TRACG and ODYN are attached in Figures I-3.3-1 thru I-3.3-3. A PCT plot is also presented for the TRACG sensitivity study. The TRACG calculated integrated SRV flow during an isolation ATWS event with reactor depressurization is bounded by the ODYN integrated SRV flow without depressurization. This conclusion is valid for all three TRACG cases corresponding to the water level controlled at TAF+5, TAF and MSCWL. [[

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[[

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Figure I-3.3-1
Integrated SRV Flow

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Figure I-3.3-2
Dome Pressure

[[

]]

Figure I-3.3-3
Peak Cladding Tempreature

NRC RAI I-3.4

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.

GE Response

The ODYN code has been previously approved by the NRC to perform BWR overpressure licensing calculations independent of the TRACG evaluation (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). Therefore, it is unnecessary to qualify the peak vessel pressure value from the ODYN code against that from the TRACG. Similarly, the ODYN/ISCOR/TASC methodology is independent of the TRACG evaluation.

Nevertheless, a comparison of the overpressure results under the same plant conditions is provided in RAI 3.3. The peak vessel pressure from ODYN calculation bounds those from the TRACG code. After the initial pressurization, the performance between ODYN and TRACG is similar. [[

]] These PCT values are comparable to the results from the ODYN/ISCOR/TASC methodology.

The integrated SRV flow from ODYN bounds those cases from TRACG assuming reactor depressurization. Therefore, the ODYN is conservative in predicting the suppression pool heatup in comparison with TRACG code. The qualification of ODYN for long-term calculation is still valid for MELLLA+ domain for all the water level control strategies.

It is concluded that the plant-specific ODYN analysis based on the TAF+5 water level strategy

would bound the TRACG analysis with or without depressurization, for the entire BWR fleet.

NRC RAI I-3.5

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)?

GE Response

The ODYN is qualified to perform long-term ATWS calculation with the following restrictions:

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. ODYN code will terminate execution once the upper plenum becomes subcooled, which is consistent with Item 2. GE calculation procedure recommends users to avoid creating this scenario during the simulation of ATWS calculation. Similarly, 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."

NRC RAI I-4.0, 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.

NRC RAI I-4.1

Provide the results of a TRACG calculation for a nonisolation 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 nonisolation ATWS with and without mitigation actions.

GE Response

A limiting ATWS instability event was simulated with operator actions recommended by the EOPs (i.e. Water level reduction to below the feedwater sparger and boron injection). The event was initialized at the [[

]] as the limiting case reported in Section 9.3.3 of NEDC-33006P Revision 1.

The water level was lowered to the top of active fuel and boron was injected from the upper plenum. Figures I-4.1-1 through I-4.1-6 show the transient response of the turbine trip with 100% bypass ATWS instability event. This TRACG simulation demonstrates that the combination of boron injection and immediate lowering of water level to below the FW spargers effectively mitigates an ATWS instability event with large amplitude power oscillations. [[

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[[

Figure I-4.1-1 Core Power and Core Flow

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[[

Figure I-4.1-2. Steam Flow, FW Flow, Water level

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[[

Figure I-4.1-3 Dome Pressure

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[[

Figure I-4.1-4. FW Temperature and Core Inlet Subcooling

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[[

Figure I-4.1-5. Limiting Bundle Power

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[[

Figure I-4.1-6. Limiting Bundle PCT

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NRC RAI I-4.2

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

GE Response

The core power and PCT responses are provided in Figures I-4.2-1 and I-4.2-2for both the explicit method (i.e. TRACG stability numeric) and the implicit method. [[
]] which eliminates the onset of large amplitude power oscillation in the explicit calculation. [[
]]

[[

Figure I-4.2-1
Reactor Power

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[[

Figure I-4.2-2
Peak Cladding Temperature

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NRC RAI I-4.3

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

GE Response

It is demonstrated that the operator actions prescribed in the EPGs are effective in managing ATWS instability concerns under MELLLA+ operating conditions.

NRC RAI I-5.0, Impact of Depressurization During ATWS Events 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.

NRC RAI 5.1

Provide detailed results of core variables during TRACG calculations for ATWS events with depressurization, including at least core and vessel void fractions, fuel temperature profiles and time evolution, boron concentrations at several elevations in the lower plenum, recirculation flow, pressure, and power levels.

GE Response

The results of the following parameters are provided in the attached figures:

Figure I-5.1-1	Reactor power
Figure I-5.1-2	Dome pressure
Figure I-5.1-3	Feedwater flow (ECCS as part of the FW system)
Figure I-5.1-4	Reactor water levels
Figure I-5.1-5	Void fraction for the bypass (Level 7).
Figure I-5.1-6	Core average void fraction
Figure I-5.1-7	SRV flow
Figure I-5.1-8	Peak Cladding Temperature
Figure I-5.1-9	Axial profiles of hot rod centerline and clad temperature.
Figure I-5.1-10	Boron flux at upper region of lower plenum [[ ]]
Figure I-5.1-11	Void fraction at upper region of lower plenum.
Figure I-5.1-12	Boron concentration at upper region of lower plenum.
Figure I-5.1-13	Recirculation flow

[[

Figure I-5.1-1
Reactor Power

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[[

Figure I-5.1-2
Dome Pressure

]]

[[

Figure I-5.1-3
Reactor Inventory Makeup Flow

]]

[[

Figure I-5.1-4
Reactor Water Level

]]

[[

Figure I-5.1-5
Core Bypass Region Void Fraction

]]

[[

Figure I-5.1-6
Average Core Void Fraction

]]

[[

Figure I-5.1-7
SRV Flow Rate

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[[

Figure I-5.1-8
Peak Cladding Temperature

]]

[[

Figure I-5.1-9a
Axial Temperature Distribution at Fuel Centerline

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[[

Figure I-5.1-9b
Axial Temperature Distribution at Outer Clad

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[[

Figure I-5.1-10a
Boron Flux in Upper Region of Lower Plenum – Ring 1

]]

[[

Figure I-5.1-10b
Boron Flux in Upper Region of Lower Plenum – Ring 2

]]

[[

Figure I-5.1-11a
Void Fraction in Upper Region of Lower Plenum – Ring 1

]]

[[

Figure I-5.1-11b
Void Fraction in Upper Region of Lower Plenum – Ring 2

]]

[[

Figure I-5.1-12a
Boron Concentration in Upper Region of Lower Plenum – Ring 1

]]

[[

Figure I-5.1-12b
Boron Concentration in Upper Region of Lower Plenum – Ring 2

]]

[[

Figure I-5.1-13
Recirculation Flow Rate

]]

NRC RAI I-5.2

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?

GE Response

The boron concentration in the upper region of the lower plenum shown in Figure I-5.1-12 of RAI 5.1 indicates that the [[

]]

NRC RAI I-5.3

Provide a series of steady-state sensitivity analyses to demonstrate that the core will remain subcritical following depressurization. Provide the core average 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.

GE Response

A sensitivity study was performed with a [[.

]] The results indicate that the core is voided after depressurization for both core flow conditions.

Core Flow (%)	[[
5	
15	]]

NRC RAI I-6.0, Containment Performance During Isolation ATWS at MELLLA+ Conditions

NRC RAI I-6.1
Provide a comparison of ODYN results of isolation ATWS simulations at MELLLA+ and original licensed thermal power (OLTP).

GE Response
The peak values from the ODYN analysis for isolation ATWS events are listed in the following table:

[[		

]]

NRC RAI I-6.2

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.

GE Response

The sequence of events for the MSIVC event is listed below:

BWR4			
Item	**Response**	**OLTP Event Time (sec)**	**M+ Event Time (sec)**
1	[[		
2			
3			
4			
5			
6			
7			
8			
9			
10			
11			
12			
13			
14			
15			]]

BWR6*			
Item	Response	OLTP Event Time (sec)	M+ Event Time (sec)
1	[[		
2			
3			
4			
5			
6			
7			
8			
9			
10			
11			
12			
13			
14			]]

* 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.

NRC RAI I-6.3

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 GENE 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 staff acknowledges that GENE may not have access to the plant-specific ATWS analysis-of-record for plants with other reload vendors.

GE Response

The peak vessel pressure and peak suppression pool temperature data for core power and flow conditions are listed in the following two tables. The impact of cycle extension from 18 months to 24 months is not included. Due to the change in methodology from REDY to ODYN, it is necessary to re-establish the baseline for the EPU or PU calculation so a consistent and meaningful comparison can be provided.

Table 1 presents the peak vessel pressure results from the ODYN methodology except those from NEDE-24222 and NEDE-24223. The results for the generic NEDE-24222 and NEDE–24223 are based on REDY methodology. The assumed rod line in NEDE-24222 and NEDE–24223 is ELLLA.

Table 2 shows the suppression pool temperature results from the ODYN/STEMP methodology except those from NEDE-24222 and NEDE-24223. The results for the generic NEDE-24222 and NEDE–24223 are based on REDY/STEMP methodology.

Table 1. Peak Vessel Pressure (psig)

[[				

]]

Table 2. Peak Suppression Pool Temperature (°F)

[[				
				]]

*N/A: Not Available

NRC RAI I-7.0, LTR Section 10.9, "Emergency and Abnormal Operating Procedures"
Section 10.9 states: "The plant EOPs will be reviewed for any effect of MELLLA+, and the EOPs updated if necessary."

NRC RAI I-7.1
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.

a. 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?

b. 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.

c. 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?

d. 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?

e. 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?

f. 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?

g. 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?

h. 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?

GE Response

a. Operator actions for control of reactor pressure during an ATWS event include steps to establish and control the pressure below 1000 psig. This is well below the maximum pressure scale shown on the HCTL. Brunswick Emergency Operating Procedures (EOPs) do not rely on SRVs cycling at setpoint but, rather, direct operator action to establish a pressure band with a high value of 1000 psig. In developing the procedure changes to support MELLLA+ implementation, this pressure was evaluated and maintained at the established criteria (1000 psig). Failure to manually control pressure and remain within the limits of the HCTL will result in emergency RPV depressurization.

b. See discussion, above, regarding the pressure established for response to the methodology for ATWS consideration. The procedure changes developed to support MELLLA+ used the corner of the MELLLA+ domain as the boundary of the operating domain.

c. The BIIT calculation was re-performed in developing the procedure changes to support MELLLA+ implementation using the established EPG/SAG methodology. The primary change related to ATWS relating to Brunswick's implementation of MELLLA+ has been to change the Standby Liquid Control solution concentration.

d. Minimum number of SRVs required for decay heat removal is based upon decay heat 10 minutes after reactor shutdown. The procedure changes developed to support MELLLA+ implementation showed that MELLLA+ has a minimal effect on the shutdown decay heat rate at the ten-minute mark after shutdown, thus not having any appreciable effect on the number of SRVs required.

e. The Minimum Number of Safety Relief Valves Required for Emergency Depressurization (MNSRED) with reactor not shutdown is based on steam flow through fuel bundle which is required to maintain fuel temperature less than 1500°F. The procedure changes developed to support MELLLA+ implementation showed that MELLLA+ has a minimal effect on this

value as it is predominately driven by fuel type and peak bundle power, thus not having any appreciable effect on the MNSRED.

f. Minimum Steam Cooling Pressure is based on the steam flow through the fuel bundle which is required to maintain temperature less than 1500° F. The procedure changes developed to support MELLLA+ implementation showed that MELLLA+ has a minimal effect on this value as it is predominately driven by fuel type and peak bundle power, thus not having any appreciable effect on this EOP value.

g. Minimum Steam Cooling RPV Water Level is also based on the steam flow through the fuel bundle which is required to maintain temperature less than 1500° F. The procedure changes developed to support MELLLA+ implementation showed that MELLLA+ has a minimal effect on this value as it is predominately driven by fuel type and peak bundle power, thus not having any appreciable effect on this EOP value.

h. Minimum Zero Injection RPV Water Level is not used in an ATWS strategy. It is used for Steam Cooling Without Injection with the reactor shutdown. It is based on the steam flow through the fuel bundle which is required to maintain temperature less than 1800° F. The procedure changes developed to support MELLLA+ implementation showed that MELLLA+ has a minimal effect on this value as it is predominately driven by fuel type and peak bundle power, thus not having any appreciable effect on this EOP value.

NRC RAI I-7.2

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.

GE Response

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.

Part II - Section 9.3.3, "ATWS with Core Instability"

NRC RAI II-1.1

Table 9-5 lists the fuel response for the set of ATWS instability analyses. Figures 9-5 to 9-11 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 in each case.

GE Response

The LTR revision will include additional information to indicate the initial power and flow, event type and mitigation strategy for the figures and the table as appropriate.

NRC RAI II-1.2
Footnote 2 to Table 9-5 states: [[

]] Please, explain this statement. [[

]]

GE Response
The ATWS instability analysis was performed to evaluate the event with mitigation and without mitigation. For both mitigated and non-mitigated cases, the event was initialized at [[

]]

NRC RAI II-1.3

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.

GE Response

The limiting ATWS instability event initiated from the MELLLA+ conditions was evaluated with TRACG. [[

]]

The original ATWS instability analysis showed that for the same event initiated from the MELLLA conditions, about 12% of the bundles in the core exceed 2200°F. It is noted due to differences in assumed analysis conditions, the two cases may not provide a valid comparison.

NRC RAI II-1.4

Please provide the results of a calculation similar to the unmitigated ATWS/stability case, but following the EOP mitigation actions. For this case, the condenser and feedwater should be assumed to be available. The purpose of this calculation is to demonstrate that the mitigation actions prescribed in the EOPs are still effective in suppressing the oscillations during operation under the EPU/MELLLA+ initial conditions. Provide a discussion of the result of this calculation.

GE Response

This is addressed in the response to RAI I-4.1.

NRC RAI II-1.5
Considering the variation that exists through the BWR fleet, explain why [[
]]

GE Response
A power to flow ratio of [[ ]]

NRC RAI II-1.6

Discuss the scoping criteria, if any, used to select the combination of limiting BWR plant physical configuration characteristics and operating parameters. [[ ]] was selected for performing the ATWS instability analyses. Include in the discussion the bases [[ ]] 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.

GE Response

ATWS instability events are evaluated to assure that the core coolable geometry criterion is met. This is accomplished by demonstrating that the PCT remains below the 2200°F limit. The ATWS instability event that most severely challenges the fuel integrity is the turbine trip with 100% bypass capacity (TTWB). TTWB completely isolates feedwater heating upon turbine stop valve closure and rapidly increases the core inlet subcooling. The core inlet subcooling reaches a maximum when the feedwater temperature reaches an equilibrium at approximately the condenser discharge temperature. As the inlet subcooling increases, the core oscillations become progressively larger and more irregular. Unlike the MSIVC event where the system response is perturbed by SRV actuations, TTWB has no external feedback to interfere with the core response and results in the largest amplitude oscillations. Since most operating BWRs have less than 100% bypass capacity, this is a conservative assumption.

The LaSalle BWR5 plant is selected for the evaluation in part due to the availability of a past ATWS instability analysis modeled with the same plant (*NEDO-32047-A, ATWS Rule Issues Relative to BWR Core Thermal-Hydraulic Stability*). [[

]]

The above assumptions provide a reasonably bounding ATWS instability event initiated from the MELLLA+ operating domain.

NRC RAI II-1.7

[[

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

GE Response

The analysis was performed with the GE14 fuel design (10x10) assumption as it is the fuel design that is currently expected to support EPU/MELLLA+ operation. However, to address transition cycles where some fuel designs other than GE14 may be expected in the core, a sensitivity study was performed to evaluate the GE13 (9x9) response to ATWS instability events. [[

]]

With the mitigation strategies recommended by the EOPs, the core coolable geometry criterion would not be compromised following an ATWS instability event assuming any GE fuel designs up to GE14.

NRC RAI II-1.8

Provide the bases and technical justifications that demonstrate that the [[ ]] 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 restriction 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.

GE Response

The fuel response to ATWS instability is dependant on the assumed fuel design. 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.

For the purpose of justifying a different vendor's fuel design or future fuel design beyond GE14, a confirmatory ATWS instability analysis should be performed with the applicable fuel and core design assumption. This analysis should be performed with TRACG (or equivalent analytical model) to simulate the limiting TTWB event resulting in regional oscillation mode.

NRC RAI II-1.9

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.

GE Response

The ATWS instability simulation did not model the lower tie plate with debris filter. Not modeling the debris filter results in a higher two-phase to single-phase pressure drop across the channel and provides a conservative condition for core instability.

ATWS evaluations without oscillations are performed on a plant specific basis and an appropriate lower tie plate assumption corresponding to the evaluated plant/cycle is used.

NRC RAI II-1.10

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, GENE recommended that hot-channel control rods not be withdrawn fully until after the pump upshift 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:

a. 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?

b. 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?

GE Response

It is true that EPU core designs result in a greater number of bundles near the maximum power as allowed by the fuel MCPR and LHGR limits. However, strictly speaking, MELLLA+ core designs should not need higher power peaking than MELLLA core designs. [[

]]

Therefore, there will not be a detrimental effect on stability margins during plant startup for MELLLA+ compared to MELLLA.

Part III - Safety Systems Actuation Limits

NRC RAI III-1

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)?

GE Response

The NPSH limits are discussed below for operations of safety systems (e.g. RHR, CS, HPCI) that take suction from suppression pool during the accident and non-accident events.

A. NPSH Limit During Accident

The NPSH limits for safety systems that take suction from suppression pool during an accident are different for plants with different vintage and their licensing commitments. In general, they 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). as described below.

Pre-Regulatory Guide 1.1 Design

[[

]]

Many of these plants now have licensing commitments, which adopted NRC imposed restrictions on NPSH design with limited or no containment overpressure credits. [[

]]

Regulatory Guide 1.1 Design

The NPSH design for plants in this category is not dependant on containment overpressure credit. It is based on 0 psig containment pressure and highest suppression pool temperature.

[[

]]

B. NPSH Limit During Non-Accident Events

[[

]]

C. NPSH Limit for HPCI pumps

The HPCI design requires HPCI system operation be limited to a suppression pool temperature of 140°F based on HPCI pump lube oil system operating temperature limits. Since 140°F is HPCI pump operational limit, it is also the NPSH limit for the HPCI pumps for both accident and non –accident events.

NRC RAI III-2

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?

GE Response

Following an initial pressure spike, the maximum reactor pressure during a limiting ATWS event is determined by the number of SRVs available and their set points. As such, this value differs from plant to plant (the 1200 psia cited in the RAI appears to have come from the Brunswick analysis for an MSIVC event at BOC).

The limiting HPCI and RCIC systems design performance requirements at any plant are based upon the set point upper analytical limit of the lowest SRV group. Therefore, the full combined HPCI and RCIC design flow rates would be available for injection, as assumed in ATWS analysis, as long as reactor pressure (following the initial spike, which decays away before any high pressure system initiation) is controlled at, or below, the upper analytical limit of the lowest SRV group set point.

It is possible that, for certain SRV OOS conditions, the peak reactor pressure during the oscillatory period of interest may exceed the maximum pressure at which the HPCI and RCIC systems are designed to deliver rated flow. If so, the effect is an insignificant reduction in the total flow injected, which is judged acceptable for the following reasons:

- The duration of the oscillatory period would be much shorter than that analyzed, as operators would undoubtedly take manual control of the SRVs to prevent their repeated cycling, and to regulate reactor pressure.

- There is conservatism in the injection requirement, in that the sum of HPCI and RCIC rated flows is assumed in the analyses; the actual flow rate required to keep the core covered is much less.

- HPCI and RCIC are constant flow systems up to the capability of the turbine controls – thereafter they would 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.

- The time spent above design pressure and, therefore, below rated flow (as described above), would be insignificant compared to the time at rated flow, such that its effect on the integrated flow volume would be likewise insignificant.

The only other system similarly affected by the subject high back pressure is RCIC, which effect and behavior is included in the discussion above. HPCS is inhibited per the EPGs, and therefore, no injection flow from HPCS is assumed for ATWS.

NRC RAI III-3
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?

GE Response
As described in Section 5.11.2 of the EPU LTR (NEDC-32424P) and Section 10.3 of the MELLLA+ LTR (NEDC 33006P) evaluations are performed to confirm that existing environmental envelopes for safety grade equipment remain valid with EPU and MELLLA+ conditions. The limiting pressures in the containment for these conditions occur for the DBA-LOCA as determined by the Task 0400 containment analyses. These pressures bound the pressures resulting from the ATWS event. The following paragraphs compare the DBA-LOCA and ATWS containment pressure response.

The maximum drywell pressures for a DBA-LOCA are significantly higher than the peak drywell pressure values for an ATWS event. This is demonstrated in Table 1 below, which shows representative calculated peak drywell pressures for the three BWR containment types (Mark I, II and III). The high drywell pressures during a DBA-LOCA are attributed to the large and rapid transfer of mass and energy into the drywell during a DBA-LOCA. (Note that the ATWS analysis calculates a peak wetwell pressure and sets the peak drywell pressure equal to the peak wetwell pressure).

The peak DBA-LOCA pressure in the wetwell will also be significantly higher for the DBA-LOCA than for the ATWS event for plants with Mark I and Mark II containments as shown in Table 1. The higher wetwell pressures during the DBA-LOCA for these plants is due to the transfer of non-condensible gas from the drywell to the wetwell, which occurs during a DBA-LOCA but does not occur during the ATWS event. The transfer of non-condensible gas adds to the wetwell pressurization caused by the increase in the wetwell airspace vapor pressure and airspace temperature increase resulting from the suppression pool heatup. For Mark I and Mark II containments, the drywell and wetwell volumes are similar in size. Consequently the transfer of drywell non-condensible gas to the wetwell during a DBA-LOCA results in a significant pressurization of the wetwell.

For Mark III containments, the peak wetwell pressure for the ATWS event and for the DBA-LOCA event will not differ significantly (Table 1 shows a difference of only 0.2 psi for the chosen representative plant). This is because the wetwell volume in a Mark III plant is much larger than the drywell volume. With a wetwell volume much greater than the drywell volume in the Mark III containment, the transfer of drywell non-condensible gas to the wetwell during a DBA-LOCA has less of an effect on the peak wetwell pressure.

Table 1 – Comparison of Peak Drywell and Wetwell Pressures DBA-LOCA vs ATWS

	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.

Part IV - Questions Related to ODYN Calculations

The staff has reviewed ODYN data for ATWS events for three plants (Brunswick, Browns Ferry, and Clinton) 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.

NRC RAI IV-1

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

]] Please answer the following questions:

a. What is the difference between the two calculations?

b. What is the applicable peak pressure limit?

c. If the applicable limit is 1500 psig, is it violated by the ODYN calculation results provided?

d. Give what the peak pressures are for other analyzed ATWS cases, including PRFO.

e. Provide the ODYN results as a function of time for the limiting ATWS event for Brunswick.

GE Response

a. The case corresponding to the peak vessel pressure of [[

]]

b. The ATWS overpressure acceptance criteria is 1500 psig.

c. The ODYN results show a comparison between EPU/MELLLA and EPU/MELLLA+ with the same SRV configuration, i.e., 1 SRV OOS. BSEP has revised the Technical Specification to require all SRV in service during the MELLLA+ operation. If one SRV is OOS, the plant must exit the M+ operating domain. This ensures that the 1500 psig limit is not violated.

d. The overpressure analysis results are listed in the table below. Since the peak vessel pressure with all SRV operational [[ ]] shows sufficient margin to the 1500 psig limit, no other comparable calculation is performed.

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

e. Figures IV-1-1 through IV-1-4 provide the plots for the PRFO/EOC with 1 SRVOOS case under EPU/MELLLA+ conditions.

[[

Figure IV-1-1 - PRFO Transient Response at LPU/MELLLA+ and EOC

]]

[[

Figure IV-1-2 - PRFO Transient Response at LPU/MELLLA+ and EOC

]]

[[

Figure IV-1-3 - PRFO Transient Response at LPU/MELLLA+ and EOC

]]

[[

Figure IV-1-4 - PRFO Transient Response at LPU/MELLLA+ and EOC

]]

NRC RAI IV-2

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.

GE Response

The two primary operator action assumed in the ODYN ATWS analysis are manual FW runback and manual boron injection. For plants equipped with Redundant Reactivity Control System (RRCS), the actual time delays for the feedwater (FW) runback and boron injection are used in the evaluation. [[

]] This is caused by the vessel isolation and not caused by the operator action.

NRC RAI IV-3

In the Brunswick calculation, the water level is raised at [[ ]] 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?

GE Response

Based on the EPU/M+ ATWS task report for Brunswick, the water level is [[

]] The following table provides the relevant sequence of events related to boron injection in the Brunswick EPU/M+ evaluation for the MSIVC case:

Item	Response	EPU Event Time (sec)	M+ Event Time (sec)
1	MSIV Isolation Initiates	[[	
2	High Pressure ATWS Setpoint		
3	BIIT Reached		
4	SLCS Pumps Start		
5	Water Level Increased		]]

[[

]]

NRC RAI IV-4

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?

GE Response

A sensitivity study was performed for the Brunswick plant at MELLLA+ conditions to determine how the ATWS suppression pool temperature is affected by the assumption of different HSBW values. [[

]] A particular HSBW value translates into the amount of time required to inject the HSBW. For example, if HSBW of 400 ppm is assumed, then the HSBW injection time proportionally reduces to 922 seconds. [[ ]] A summary of results is presented in Table 1.

[[

]]

Table 1. Sensitivity Results – Summary

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] [[

]]

[[

]]

Figure IV-4-1. ODYN Core Average Boron Concentration Response

NRC RAI IV-5

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.

GE Response

The boron mixing and remixing efficiency in the ODYN code is a function of core flow.
[[

]] so that the resulting total SRV discharge to the suppression pool bounds that from the TRACG calculation. The core flow threshold for boron mixing efficiency is disclosed in Section 4.2.2.2 in ODYN qualification LTR (NEDC-24154P-A).

NRC RAI IV-6

Section 9.3.1 of the Brunswick MELLLA+ LTR (NEDC-33063P) states that the MELLLA+ analysis was performed with 10-percemt 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.

GE Response

The ATWS overpressure evaluation was performed for the bounding [[ ]] case with a special SRV setpoint assumption. The SRV opening setpoint drift allowance is set to 3% for all the valves except for the one valve in the lowest setpoint group. [[ ]] The resulting valve opening setpoints are:

Valve Group	Opening Setpoint (psig) for 1SVOOS Evaluation	Opening Setpoint (psig) for 0SVOOS Evaluation
1	[[	
2		
3		
4		
5		
6		
7		
8		
9		
10		
11		

]]

The nameplate capacity of 829,000 lbm/hr at a reference pressure of 1080 psig is used in the Brunswick EPU/MELLLA+ ATWS evaluation. This is the certified capacity of the Target Rock valves installed in Brunswick plants.

NRC RAI IV-7

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?

GE Response

Tables 1 and 2 provide the event sequence for an MSIVC event and a PRFO event respectively. The assumed average SRV opening setpoints are [[ ]] for the 3 valve groups. Table 3 provides the SRV opening timing for an MSIVC event and a PRFO event. It is noted that the difference in opening times among the valves in the same setpoint group is due to the implementation of statistical spread of the opening setpoints.

Table 1. Event Sequence 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 2. Event Sequence for a 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 3. 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		

Valve Group	MSIVC Event	PRFO Event
8		
9		
10		
11		]]

Part V - Clinton Specific Questions

The reference analysis for the EPU/MELLLA+ plant [Clinton] specific calculation (NEDC-33057P) states that the ATWS suppression pool temperature limit is 185 (see table in Section 9.3.1 of NEDC-33057P).

NRC RAI V-1

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.

GE Response

These RAIs related to the plant-specific application made by the Clinton (M+SAR). It is more appropriate that Clinton should make the responses to these RAIs, if necessary, on their own docket.

However, GE believes that additional information has been provided herein, which may make these Clinton RAIs unnecessary. Please refer to RAIs Parts I-3.3 and 5.1 regarding the peak suppression pool temperature and RAI I-7.1 regarding the EOPs.

NRC RAI V-2

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.

GE Response

Please see the response to RAI V-1

NRC RAI V-3

Provide the assumptions used in the ATWS analysis for the EPU/MELLLA+ pilot plant [Brunswick] specific calculation (NEDC-33063P). Specifically, what type of ATWS transient is limiting? What are the initial conditions, including power, flow, 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.)?

GE Response

The key assumptions used in the Brunswick specific M+ calculation are as follows:

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 V-4

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.

a. 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?

b. 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.

c. 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 mininum of SRVs does change, will this affect any other variables?

d. 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?

e. 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?

f. 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?

g. 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?

GE Response
Please see the response to RAI V-1