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Constellation Energy
Nine Mile Point Nuclear Station

October 16, 2007

U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

ATTENTION: Document Control Desk

SUBJECT: Nine Mile Point Nuclear Station
Unit No. 2; Docket No. 50-410

Response to Request for Additional Information:
Implementation of ARTS/MELLLA (TAC No. MD5233)

REFERENCES:

(a) Letter from K. J. Nietmann (NMPNS) to Document Control Desk (NRC), dated March 30, 2007, License Amendment Request Pursuant to 10 CFR 50.90: Implementation of ARTS/MELLLA

(b) Letter from M. J. David, (NRC) to K. J. Polson (NMPNS), dated August 16, 2007, Request for Additional Information Regarding Nine Mile Point Nuclear Station, Unit No. 2, Implementation of ARTS/MELLLA (TAC No. MD5233)

Pursuant to 10 CFR 50.90, Nine Mile Point Nuclear Station, LLC, (NMPNS), requested in Reference (a) approval of an amendment to the Nine Mile Point Unit 2 Renewed Operating License NPF-69 to reflect an expanded operating domain resulting from the implementation of Average Power Range Monitor/Rod Block Monitor/Technical Specifications/Maximum Extended Load Line Limit Analysis (ARTS/MELLLA). The purpose of this letter is to provide responses to the request for additional information (RAI) transmitted to NMPNS in Reference (b).

Non-proprietary responses to the RAI are provided in Attachment (1). A proprietary version of the responses is provided in Attachment (3). Certain information in Attachment (3) is considered by General Electric-Hitachi Nuclear Energy Americas LLC (GEH) to be proprietary information exempt from disclosure pursuant to 10 CFR 2.390. Therefore, on behalf of GEH, NMPNS hereby makes application to withhold Attachment (3) from public disclosure in accordance with 10 CFR 2.390(b)(1). An affidavit executed by GEH detailing the reasons for the request to withhold the proprietary information is provided in Attachment (2).

This response does not affect the No Significant Hazards Determination analysis provided by NMPNS in Reference (a). Pursuant to 10 CFR 50.91(b)(1), NMPNS has provided a copy of this response, with the non-proprietary attachment, to the appropriate state representative.

Should you have any questions regarding this submittal, please contact T. F. Syrell, Licensing Director, at (315) 349-5219.

Very truly yours,

Keith J Polson

STATE OF NEW YORK :
: **TO WIT:**
COUNTY OF OSWEGO :

I, Keith J. Polson, being duly sworn, state that I am Vice President-Nine Mile Point, and that I am duly authorized to execute and file this response on behalf of Nine Mile Point Nuclear Station, LLC. To the best of my knowledge and belief, the statements contained in this document are true and correct. To the extent that these statements are not based on my personal knowledge, they are based upon information provided by other Nine Mile Point employees and/or consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.

Keith J Pol
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Subscribed and sworn before me, a Notary Public, in and for the State of New York and County of Oswego, this 16th day of october, 2007.

WITNESS my Hand and Notarial Seal:

Sandra A Oswald
Notary Public

My Commission Expires:

SANDRA A. OSWALD
Notary Public, State of New York
No. 01OS6032276
Qualified in Oswego County
Commission Expires 10-25-09

10/25/09
Date

KJP/JJD

Attachments: (1) Response to Request for Additional Information Regarding Implementation of ARTS/MELLLA - Non-Proprietary Version
(2) GEH Affidavit
(3) Response to Request for Additional Information Regarding Implementation of ARTS/MELLLA – Proprietary Version

cc: M. J. David, NRC
S. J. Collins, NRC (without Attachments 2 and 3)
Resident Inspector, NRC (without Attachments 2 and 3)
J. P. Spath, NYSERDA (without Attachments 2 and 3)

ATTACHMENT (1)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

REGARDING IMPLEMENTATION OF ARTS/MELLLA

NON-PROPRIETARY VERSION

Certain information, considered proprietary by GEH, has been deleted from this Attachment. The deletions are identified by double square brackets.

By letter dated March 30, 2007, Nine Mile Point Nuclear Station, LLC, (NMPNS) submitted a license amendment request (LAR) for Nine Mile Point Unit 2 (NMP2) Renewed Operating License NPF-69. The proposed amendment would reflect an expanded operating domain resulting from the implementation of Average Power Range Monitor/Rod Block Monitor/Technical Specifications/Maximum Extended Load Line Limit Analysis (ARTS/MELLLA). The Average Power Range Monitor (APRM) flow-biased simulated thermal power scram Allowable Value would be revised to permit operation in the MELLLA region. The current flow-biased Rod Block Monitor (RBM) would also be replaced by a power dependent RBM which also would require new Allowable Values. In addition, the flow-biased APRM simulated thermal power setdown requirement would be replaced by more direct power and flow dependent thermal limits to reduce the need for manual APRM gain adjustments and to provide more direct thermal limits administration during operation at other than rated conditions.

The NRC issued a request for additional information (RAI) concerning the NMP2 license amendment request for implementation of ARTS/MELLLA on August 16, 2007. The NMPNS responses to the RAI questions follow.

NRC Question 1

On page 4-12 of Attachment (7) of your request, it was stated that the performance of the system was upgraded such that the rod withdrawal error (RWE) event will never be the limiting transient. The RWE transient minimum critical power ratio (MCPR) is determined by the rod block monitor (RBM) setpoints. These setpoints will be selected based on the operating limit minimum critical power ratio, as established by other anticipated operational occurrences (AOOs), and the RBM setpoints will remain in the TS. The NRC staff understands that in the event the setpoints are exceeded due to failure of RBM, then the RWE will violate the safety limit minimum critical power ratio (SLMCPR). If that is the case, then why should the RBM not be treated similar to other safety related systems, and be classified as such? Please explain.

NMPNS Response 1

The classification of the RBM as non-safety related system is consistent with the RBM's generic classification for all General Electric-Hitachi Nuclear Energy Americas LLC (GEH) boiling water reactors (BWRs).

The classification of the RBM system was most recently documented in the General Electric Digital Nuclear Measurement and Control Power Range Monitor (NUMAC PRNM) Licensing Topical Report (NEDC-32410 P-A), which was approved by the NRC. Section 3.3 of the NRC safety evaluation report (SER) states that the RBM chassis is not required to operate in order to accomplish a system safety function. Section 3.3.2 lists the safety functions of the PRNM system. The listed functions only include the APRM and oscillation power range monitor (OPRM), which do not include the RBM. Also, the NRC's SER states that "There are no safety functions associated with the rod block or information output."

The regulatory basis for this classification is found in 10 CFR 50.2 which defines safety related structures, systems, and components (SSCs) as those SSCs that are relied upon to remain functional during and following design basis events to assure:

(1) integrity of the reactor coolant pressure boundary,

(2) capability to shutdown the reactor and maintain the reactor in a safe shutdown condition, or

(3) prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in 10 CFR 50.34(a)(1) or 10 CFR 100.11.

The RBM is a system that mitigates the consequences of a RWE by automatically initiating a rod block to ensure that the SLMCPR is not exceeded. The RWE is not an accident. It is an AOO, which, as defined in 10 CFR 50, Appendix A, is a condition of normal operations. A RWE does not challenge the integrity of the reactor coolant pressure boundary, and thus, the RBM is not used to maintain the integrity of the reactor coolant pressure boundary.

The RBM has no safe shutdown function, and cannot prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in 10 CFR 50.34(a)(1) or 10 CFR 100.11.

A complete RBM failure would require more than one failure and would not be considered to be an AOO. The RBM system basis is limited to consideration of single control rod withdrawal errors and does not accommodate multiple errors. The RWE analysis basis is established to conservatively credit the most limiting of the two independent RBM channels. Section 4.5 of Attachment (7) of the LAR documents the power-dependent MCPR requirements when operability of the RBM is required to protect the SLMCPR. Cycle specific calculations are performed to confirm these requirements. In addition, these cycle specific calculations are performed to show that the fuel thermal mechanical requirements are met.

Further, the RBM is highly reliable and a high quality system that is designed to criteria and standards identical in many ways to safety related portions of the PRNM system. The RBM includes redundancy features, fail-safe features and self-monitoring features. The response to RAI Question 22e in a subsequent submittal will provide additional information regarding the quality standards applicable to the RBM.

In conclusion:

- The RWE event is not a threat to the reactor coolant pressure boundary. Thus, the RBM does not meet the first criteria of safety related SSCs (assure integrity of the reactor coolant pressure boundary).

- The RBM is not capable of shutting down the reactor and maintaining the reactor in a safe shutdown condition. Thus, the RBM does not meet the second criteria of safety related SSCs (assure capability to shutdown the reactor and maintain the reactor in a safe shutdown condition).

- The RWE is not an accident, and the RBM has no function used to prevent or mitigate the radiological consequence of any accident. Thus the RBM does not meet the third criteria of safety related SSCs (prevent or mitigate the consequences of accidents).

- The non-safety related classification of the RBM has been reviewed and approved by NRC.

Based on the above, the RBM does not meet the regulatory definition of a safety related SSC.

NRC Question 2

Regarding the turbine trip with no bypass (TTNBP), load rejection with no bypass (LRNBP), and main steamline isolation valve (MSIV) closure with a flux scram (MSIVF) transients:

NRC Question 2a

Following the initiating event for the TTNBP and LRNBP transients at NMP2, describe how the sequence of reactor protection system initiation differs between the two transient events? At NMP2, which of these two events is limiting, and why?

NMPNS Response 2a

The sequence of events for the Reactor Protection System (RPS) is provided in the table below. There is only a very small difference in the delay due to the sensing logic for the two different types of valve closures. These events are very similar, but a small difference in these events is the valve closure times. The full closure time for the turbine stop valve(s) (TTNBP) is 100 msec and the full stroke closure time for the turbine control valve(s) is 110 msec. The delta critical power ratios (ΔCPRs) for these two events are very close such that they are basically the same severity. There is some small variation in results for different initial conditions such that the TTNBP is slightly more limiting for some conditions and the LRNBP is slightly more limiting for other conditions. Given the similarity in the RPS delays and valve closure times, the two events are expected to be nearly identical.

Event	TTNBP Time (sec)	LRNBP Time (sec)
Scram Signal Generated	0.02	0.03
RPS Logic Process	0.07	0.08
Scram Solenoid Actuation	0.27	0.28

NRC Question 2b

TTNBP, LRNBP and MSIVF - all three of these transients are pressurization events. The TTNBP and LRNBP events need to be evaluated for MELLLA operation; whereas, in Section 5 of Attachment (7) of your request, it was stated that ARTS/MELLLA does not affect the vessel overpressure protection analysis. Please explain.

NMPNS Response 2b

The limiting event for the vessel overpressure analysis is the MSIVF as described in Section 5.0 of Attachment (7) of the LAR. The MSIVF results are primarily [[
]] associated with the cycle specific core design. A demonstration was provided in Table 5-1 that shows that the increased core flow condition (105% core flow) produces the more limiting peak vessel pressure for NMP2. The higher initial core flow has a higher core pressure drop and a higher initial pressure in the lower plenum and results in higher peak vessel pressures. The intention of the sentence referenced in NRC Question 2b was to indicate that the MELLLA initial condition does not adversely affect the peak vessel pressure.

NRC Question 2c

Considering all the pressurization events at NMP2, which transient is most limiting; and at what operating domain?

NMPNS Response 2c

As described in the response to NRC Question 2b above, the MSIVF is the limiting event for the ASME overpressure transient. Note that for the ASME calculation, the MSIVF includes an additional failure in the RPS system and is therefore not an AOO where MCPR is calculated.

Pressurization event thermal limit results are provided in Table 3-3 of Attachment (7) of the LAR. The ΔCPR for the LRNBP and TTNBP are nearly identical for both increased core flow (105% core flow) and MELLLA (80% core flow) initial conditions. These are the limiting pressurization transients. Table 3-3 shows that the increased core flow domain produces the more limiting ΔCPR results.

NRC Question 3

In Table 7-2 of Attachment (7) of your request, two licensing basis peak cladding temperature (PCT) values were reported - current licensing basis PCT as 1370°F, and updated PCT as 1480°F, both for a 0.07 ft^2 small break loss-of-coolant accident (LOCA). Provide the following additional information:

NRC Question 3a

Which of these two analyses (and the resulting PCT) is considered as the analysis of record?

NMPNS Response 3a

The current analysis of record for NMP2 is at rated conditions and the limiting case is identified as a design basis accident (DBA) maximum recirculation line break with a high pressure core spray (HPCS) diesel generator (DG) failure. The Appendix K PCT for this case was 1365°F which resulted in a Licensing Basis PCT of 1370°F. This analysis was updated by 10CFR50.46 notification #2006-01, which accounted for the effect of a top peaked power shape in the small break analysis and increased the Licensing Basis PCT for NMP2 by 90°F, based on an analysis for a similar plant, i.e., for a total of 1460°F (Letter from G. Harland, NMPNS, to NRC Document Control Desk, dated January 11, 2007, Report of Changes or Errors Discovered in the Current Acceptable Emergency Core Cooling System Evaluation Models). For the NMP2 ARTS/MELLLA evaluation, the limiting cases were re-analyzed including the top peaked power shape small break case. From this NMP2 plant specific calculation, the limiting case was identified as the 0.07 ft^2 recirculation line break. The Appendix K PCT for this case was 1478°F which resulted in a Licensing Basis PCT of 1480°F. This will be the new analysis of record after implementation of ARTS/MELLLA.

NRC Question 3b

What was the basis for 20°F higher feedwater temperature used for the updated analysis?

NMPNS Response 3b

The nominal heat balance for NMP2 under current conditions assumes a feedwater temperature of 423°F. The "Current Analysis" was performed at 403°F to take into account the plant allowance for -20°F feedwater temperature below the rated value, and to make this assumption consistent with feedwater assumption for other transients. The PCT results for the updated analysis have confirmed the bounding condition for emergency core cooling system (ECCS)-LOCA. The updated analysis was performed at the nominal feedwater temperature of 423°F.

NRC Question 3c

What was the licensing basis PCT at the rated power and MELLLA flow (100 percent current licensed thermal power (CLTP)/80 percent rated core flow (RCF)), which is state point E in Figure 1-1 of Attachment (7) of your request?

NMPNS Response 3c

The calculated Appendix K PCT for Point E is 1470°F, compared to 1478°F for the limiting statepoint at rated flow and rated power assuming the limiting break case. The licensing basis PCT is only calculated at the state point for the limiting Appendix K PCT. The process for calculating licensing basis PCT and the associated uncertainties are not sensitive to state point and therefore need only be applied to the limiting power/flow point as calculated for Appendix K PCT. For NMP2, the plant variable uncertainties cause a 2°F increase in the Appendix K PCT (1478°F) such that the licensing basis PCT is 1480°F. If the same 2°F increase were applied to the Appendix K PCT of 1470°F at state point E, the licensing basis PCT would be estimated to be 1472°F.

NRC Question 3d

Describe the changes made to the current analysis, and the basis for the change, in order to obtain the results for the updated analysis.

NMPNS Response 3d

The significant analysis assumption differences between the current analysis and the updated analysis are the feedwater temperature assumption [see response to NRC Question 3b] and the consideration of a top peaked axial power shape for the small break cases [see response to NRC Question 3a].

NRC Question 4

As one travels from state points F towards E along the higher MELLLA load line shown in Figure 1-1 of Attachment (7) of your request, power-to-flow ratio and core inlet subcooling increase. As a result, two competing phenomena, i.e., the time of boiling transition and core recovery time, affect the limiting PCT. The NRC staff believes that it is possible that the limiting PCT can occur somewhere between points F and E, depending on how these competing phenomena play out. Therefore, in order to confirm the limiting PCT for MELLLA operation at NMP2, provide a PCT value for the mid-point between F and E. Otherwise, please provide justification for not calculating a mid-point PCT.

NMPNS Response 4

The sensitivity of the competing effects described is small for NMP2. There is an 8°F difference in the PCT for the DBA break between the evaluation at point F and E (The point F PCT is higher than Point E by 8°F.) This effect is small for NMP2 because early boiling transition is assumed to occur for the high-powered node for the case of rated core flow (Point F). Therefore, any reduction in flow does not further impact early boiling transition of the high-powered node. The 8°F difference is an indication of the insignificance of the effect on core uncovery due to the change in core flow compared to the NMP2 Licensing Basis PCT margin to the 2200°F limit (>700°F).

The same holds true for the small break. The sensitivity of PCT to flow for the small break is also 8°F (Point F PCT is higher than Point E). Since early boiling transition is not an issue for small breaks, again the 8°F difference is an indication of the insignificance of the effect on core uncovery due to the change in core flow compared to the NMP2 Licensing Basis PCT margin to the 2200°F limit (>700°F).

NRC Question 5

Describe your training program for the operators in preparation for implementing the ARTS/MELLLA operation at NMP2.

NMPNS Response 5

Training is being developed in accordance with the Initial License and Licensed Operator Requalification Training Programs. A systematic approach to training development and implementation is being used. Vendor support was used to help develop the training that will be delivered to the licensed operators.

GEH has conducted a training session for certain NMP2 plant personnel at the site. The attendees included an Operations training instructor (tasked with developing the training modules for the plant operators), Fuels, and Reactor Engineering personnel.

The GEH training session included discussion and material on the following topics:

- Background on Thermal Limits and the Operating Power/Flow Map
- Current Licensing Criteria
- ARTS Limits Development
- Operational Impact of ARTS with respect to the Rod Block Monitor

- MELLLA Evaluation
- Technical Specifications (TSs)
- Impact of the Revised Limits on Operating Margins
- Expanded Use of the Operating Power to Flow Map
- Review of Previous Startups/Power Ascensions

Licensed Operators are scheduled to receive training on the ARTS/MELLLA modification prior to implementation.

NRC Question 6

On page 7 of Attachment (1) of your request, it is stated that the anticipated transient without scram (ATWS) analysis resulted in a peak upper plenum pressure that is 5 pounds per square inch greater than the current analysis. It was further stated that the increase in peak upper plenum pressure is not due to implementation of MELLLA, but rather to differences in the modeling assumptions used in the revised ATWS analysis based on a new One Dimensional Core Transient (ODYN) model. Provide the following additional information:

NRC Question 6a

What necessitated use of a new ODYN ATWS model? Was that because the current model was not adequate and not acceptable?

NMPNS Response 6a

The current analysis basis is with the older REDY model. As REDY models are no longer maintained, and the ODYN model is NRC approved, the ODYN model was used in the analysis. Note, the ODYN model is used for the NMP2 reload licensing calculations. The 5 psi upper plenum pressure difference stated is the small difference between REDY and ODYN calculations.

NRC Question 6b

Did the NRC staff review and approve the new model?

NMPNS Response 6b

The staff has reviewed and approved ODYN for application to ATWS in: "Qualification of the One-Dimensional Core Transient Model (ODYN) for Boiling Water Reactors," NEDC-24154P-A (Supplement 1 - Volume 4), February 2000.

NRC Question 7

On July 24, 2003, NMP2 experienced a failure of a power supply which lead to the concurrent failure of the steam flow, recirculation, and level control systems and subsequently resulted in a feedwater pump and a recirculation pump runback and downshift. The transient was terminated by an oscillation power range monitor (OPRM) SCRAM.

NRC Question 7a

Demonstrate that the OPRM setpoints in the Option III stability solution will provide adequate protection against exceeding specified acceptable fuel design limit (SAFDLs) by performing an analysis of the same initiating event starting from the limiting point in exposure from the most limiting point on the MELLLA power/flow map. Specifically address the consequences of regional mode oscillations for the planned first ARTS/MELLLA cycle core and anticipated operating strategy.

NMPNS Response 7a

A TRACG analysis was performed for NMP2 simulating regional oscillations following a two-recirculation pump trip (2RPT) event initiated at 100% rated power and minimum rated core flow (80% rated core flow) along the MELLLA boundary, which is the most limiting point on the power/flow map for initiating the 2RPT event. The TRACG model included simulation of the actual OPRM cells based on the installed NMP2 OPRM system, which uses the 4P design with four local power range monitors (LPRMs) per OPRM cell. [[ ]] This simulation and the results discussed below bound the July 24, 2003 event, which included an unplanned flow runback and recirculation pump downshift on the ELLLA rod line.

The analysis is performed by using TRACG04A and PANAC11A computer codes. [[ ]] The same exposure statepoint was used in the Option III evaluation for NMP2 ARTS/MELLLA DIVOM application. It should be noted that analysis conditions (e.g., rod pattern, exposure, OLMCPR) performed in the evaluation are [[ ]]

The OPRM cell responses are passed through the Period Based Detection Algorithm (PBDA) to determine the time of trip by evaluating the number of Successive Confirmation Counts (SSC) and signal amplitude, S_p. [[ ]]

A 2RPT event is a slowly evolving transient with minimal system interaction. The 2RPT event is simulated by manually tripping the pumps from a desired steady state operating condition followed by a slow runback. The pump trip initiates a core flow coast down to natural circulation flow. [[

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Using the trip time determined from the OPRM response [[

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This result demonstrates that OPRM setpoints in the Option III stability solution provide substantial margins against exceeding the safety limit MCPR for regional mode oscillations, such as that which occurred during July 24, 2003. This TRACG04 simulation demonstrates the substantial conservatisms inherent in the Option III DIVOM-based licensing basis methodology.

Should the event generate no oscillations or weak oscillations that do not reach the trip setpoint after the initial runback, the operating strategy is to move out of the region immediately by inserting control rods and/or increasing the core flow. This will prevent a thermal-hydraulic instability event driven partly by the feedwater temperature, which reaches a lower equilibrium value a few minutes into the transient.

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Figure 7a-1: 100% Power / 80% Flow Results

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Figure 7a-2: 100% Power / 80% Flow, Channel Power Results

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Figure 7a-3: 100% Power / 80% Flow, Channel Flow Results

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Figure 7a-4: 100% Power / 80% Flow, Channel CPR Results

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Figure 7a-5: 100% Power / 80% Flow, Leading OPRM Cell Results

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Figure 7a-6: 100% Power / 80% Flow, 2RPT MCPR Results

NRC Question 7b

The OPRM armed region of the power/flow map was generically defined. The ARTS/MELLLA operating domain, however, allows for operation at certain powers off the rod line for that power. Provide an analysis of the core and channel decay ratio at a few points along the OPRM armed region boundary using an approved NRC method (such as ODYSY). Use the "dog-bite correlation" to draw conclusions regarding susceptibility to regional mode oscillations. Based on the analysis, discuss (1) any conservatism in the selection of the OPRM armed boundary, and (2) the impact of rod patterns off the rod line associated with that power on core wide stability.

NMPNS Response 7b

The OPRM Armed Region is generically defined as ≥30% rated core power and ≤60% rated core flow on the power/flow map in the Option III licensing topical report, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications," NEDO-32465-A, August 1996. Based on experience with actual instabilities and decay ratio (DR) calculations, instabilities above 60% rated core flow are not expected. Similarly, instabilities occurring below 30% rated power are also not expected. However, if an instability were to occur below 30% rated power, the instability would not be expected to grow large enough to threaten the SLMCPR. This expectation is due, in part, to the large MCPR margin that exists at low power.

Core and channel decay ratios at various points along the generically defined OPRM Armed Region boundary are provided in Table 7b-1 and plotted on Figure 7b-1 to demonstrate their relationship to the dog-bite acceptance correlation. The points on the 60% flow line were selected to show the impact on stability of the increase in operating domain associated with the implementation of MELLLA. These decay ratios are calculated using the NRC approved ODYSY code (Licensing Topical Report, "ODYSY Application for Stability Licensing Calculations," NEDC-32992P-A, July 2001). The analysis completed for this RAI Response used the Nominal Natural Circulation Line (identified as Low Recirc Pump Speed Both FCVs Min. Pos. in Figure 6-2 of Attachment (7) of the LAR) and demonstrates that the backup stability protection (BSP) boundaries shown in Figure 6-2 are conservative.

The calculated channel decay ratios for the points shown are well below 0.56, which is the criterion used to establish susceptibility to regional oscillations. In addition, the core decay ratios are significantly below the 0.80 acceptance criteria for core wide oscillations. Based on these low decay ratios, it is highly unlikely that either regional or core wide mode oscillations will occur outside the OPRM Armed Region boundary near the points shown.

Conservatisms used in this analysis include:

- [[

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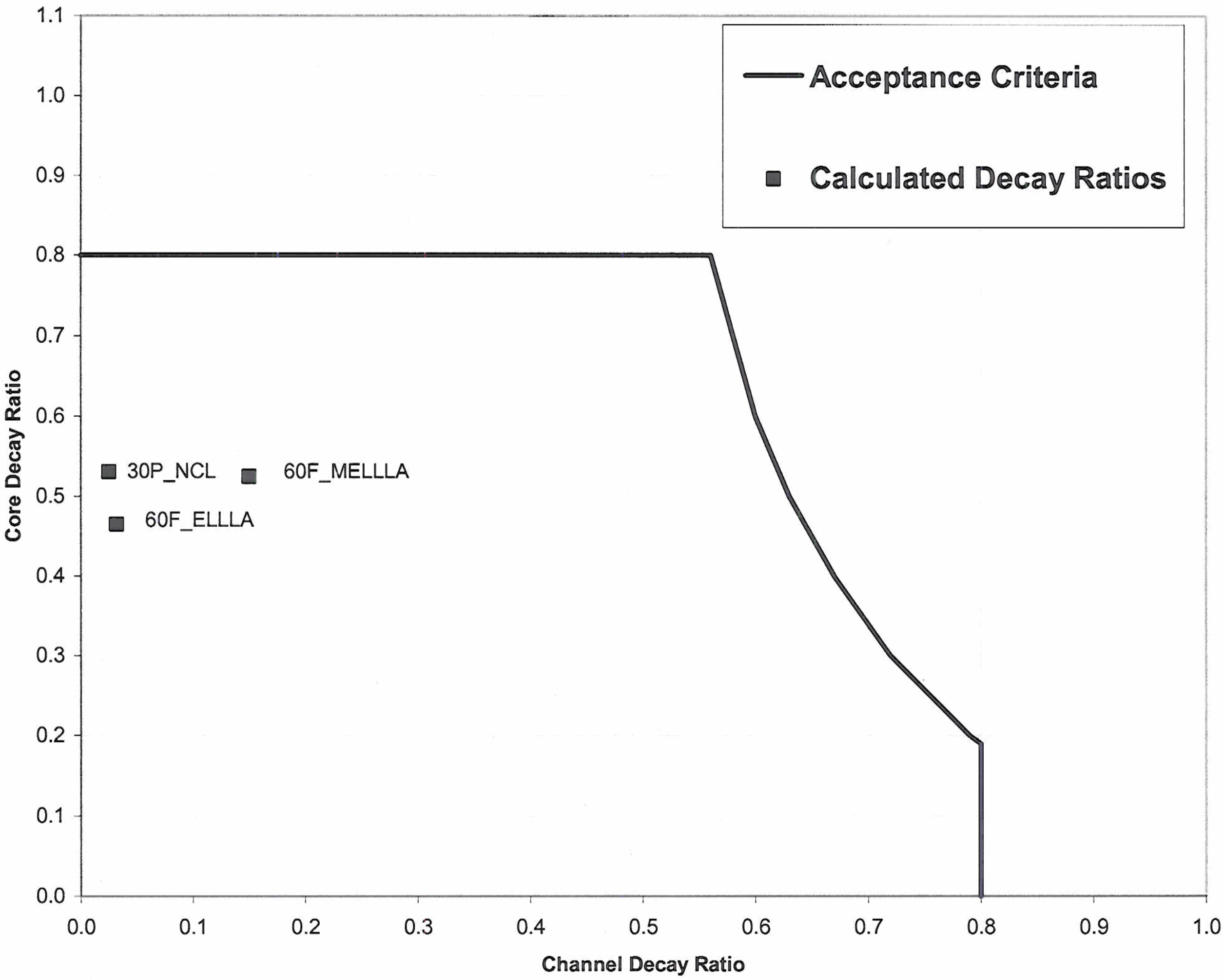
Table 7b-1
Calculated ODYSY Decay Ratios along the Generic OPRM Armed Region Boundary

Case Name	Conditions	Power (% Rated)	Flow (% Rated)	Core DR	Highest Channel DR
60F_MELLLA	Intersection of High Flow Control Line and 60% rated core flow	83.10	60.00	0.525	0.150
60F_ELLLA	Intersection of ELLLA line and 60% rated core flow	71.7%	60.00	0.465	0.032
30P_NCL	Intersection of Natural Circulation Line and 30% rated power	30.00	28.70	0.492	0.044

Figure 7b-1
Calculated ODYSY Decay Ratios and Acceptance Criteria



NRC Question 8

Regarding limiting core-wide AOOs:

NRC Question 8a

Section 3.1 of Attachment (7) of your request describes the limiting transients considered for the plant specific MELLLA application. The section refers to generic assessments of several transients. Describe the salient design features of the boiling water reactor (BWR)/5 considered in the assessments. If a specific BWR/5 was considered as part of the assessment, provide either the plant name or a brief description of the design differences between the plant considered in the assessment and NMP2.

NMPNS Response 8a

The generic assessments were performed to determine the most limiting transients and characteristics for the BWR fleet. This was done by using the plant characteristics from the fleet of BWR/3 through BWR/5 plants that resulted in the most limiting transients. The plants were chosen to cover a wide range of conditions and characteristics including steam line volume, plants with and without the recirculation pump trip feature, high and low feedwater runout capacity, and low bypass capacity. None of the BWR/5 plants had plant characteristics that were limiting for the fleet.

The key plant characteristics considered for off-rated limits calculations include:

- Steam Line Characteristics
- Feedwater (FW) Runout Capacity
- High Pressure Core Injection (HPCI) Flow Capacity (System Not Applicable to NMP2)
- Recirculation Pump Trip
- Steam Bypass Capacity
- Relief Capacity
- Design Conditions (Power Density, FW temperature, etc…)

As noted above, the only key plant characteristic considered that is not applicable to NMP2 is HPCI flow capacity, since the BWR/5 does not have a HPCI system. To confirm the applicability of the generic assessment to NMP2, plant specific calculations were performed which included all of the key plant characteristics described above that applied to NMP2. These analyses were performed with the latest approved methods [see Table 1-1 of Attachment (7) to the LAR] and the most recent core designs. These analyses confirmed the applicability of the generic assessments for the limiting AOOs to NMP2.

NRC Question 8b

What is meant by Option A or Option B in Section 3.0?

NMPNS Response 8b

Option A and Option B correspond to the use of different scram speeds. The Option A scram speed is the TS scram speed. Option B is an improved scram time that is validated by plant measurement. Option B is described in GESTAR II, General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A-15, September 2005, Section S.5.1.5.2.

NRC Question 8c

Section 3.1 of Attachment (7) of your request states that: "The LFWH [loss of feedwater heating] event is not limiting for NMP2 and the effect of MELLLA on the LFWH severity is sufficiently small that the LFWH remains non-limiting for MELLLA...considering that the LFWH event becomes less limiting as the power decreases (less feedwater to be affected by loss of heating), the LFWH event was not considered in the determination or validation of the off-rated limits." Provide the results of LFWH transient calculations starting from 100 percent CLTP and at 105 percent RCF at beginning-of-cycle. Compare the transient change in thermal margins to those for rapid pressurization events.

NMPNS Response 8c

The LFWH transient was recalculated at 100% power / 105% core flow at beginning-of-cycle (BOC) conditions. The ΔCPR was 0.14, which is considerable lower than the pressurization event results documented in Table 3-3 of Attachment (7) of the LAR. The limiting pressurization event has a ΔCPR of 0.37 with the Option B scram time.

NRC Question 8d

Describe the potential consequences of an inadvertent HPCS initiation at end-of-cycle (EOC) conditions at the 100 percent CLTP/80 percent RCF point in the MELLLA operating domain.

NMPNS Response 8d

The inadvertent HPCS initiation at EOC results in the injection of cold water in the upper plenum area above the core. This results in a small depressurization and core power decrease as some of the steam generated by the core is quenched. The pressure regulator responds to maintain the pressure at the pressure setpoint and the feedwater control system responds to the increased inventory provided by the HPCS system. The system would settle to a new steady state without a scram in this scenario with increased margins to thermal limits compared to the initial conditions due to the decreased power.

NRC Question 9

Verify that the first ARTS/MELLLA cycle core will be comprised of only General Electric fuel bundles.

NMPNS Response 9

The first ARTS/MELLLA cycle core will be comprised of only GE-14 fuel bundles.

NRC Question 10

Regarding the ATWS event:

NRC Question 10a

The standby liquid control system (SLCS) relief valve setpoint is the same as the design pressure. The NRC staff does not understand how the relief valve will protect the SLCS unless the relief valve setpoint pressure is sufficiently lower than the design pressure to allow the flow of the full SLCS injection flow rate from the pump discharge to the relief valve. Explain the claim that the revised relief valve setpoint will continue to ensure compliance with Section III of the ASME Boiler and Pressure Vessel code.

NMPNS Response 10a

The applicable Design Code of record for the NMP2 SLCS is the ASME Boiler and Pressure Vessel Code, Section III, Subsection NC, 1974 Edition.

Section NC-7511 of the 1974 Edition of the ASME code, "Set Pressure Limitations," states:

"The set pressure of at least one of the pressure relief devices connected to the system shall not be greater than the maximum allowable working pressure of the system at design temperature which it protects. Additional pressure-relief devices, other than safety relief and liquid relief valves, may have higher set pressures, but in no case shall the set pressures be such that the total accumulated pressure exceeds 110% of the system design pressure."

The maximum allowable working pressure refers to the capability of the piping system components based on actual wall thickness and component ratings, versus minimum thickness requirements under design conditions. By definition, the allowable working pressure must be equal to, or greater than design pressure.

Based on the above, the liquid relief valves on the discharge of each SLCS pump may be set to system design pressure, or 1,400 psig.

Similarly, Paragraph NC-7411 states:

"The total rated relieving capacity of the pressure relief devices intended for overpressure protection of the system whose components are within the scope of this subsection shall be sufficient to prevent a rise in pressure of more than 10% above system design pressure at design temperature within the protected boundary of the system under any pressure transients anticipated to arise. The system design pressure established for the pressure retaining boundary of the system for which overpressure protection is provided shall not exceed the design pressure of any component within the protected boundary."

Based on the above, it is common system design practice to set system relief valves at design pressure, provided the total accumulated pressure remains within the allowed 10% above design pressure.

For the SLCS pressure boundary protected by the subject relief valves, the design pressure of the system is 1,400 psig (note: Design pressure of all components within the protected boundary is at least 1,400 psig). A review of the total accumulated pressure confirmed that the pressure will remain below the maximum pressure of 1,540 psig (1.1 x 1,400 psig).

The NRC staff has previously reviewed a similar setting of relief valves at the design pressure of the SLCS for Susquehanna (TAC Nos. MC3305 and MC3306).

NRC Question 10b

Boiling transition was not considered as a fuel integrity acceptance criterion. Describe the consequences of a non-isolation ATWS initiated by an inadvertent dual recirculation pump trip from the 100 percent CLTP/80 percent RCF operating point at EOC. Determine the limiting non-isolation ATWS event, considering two-loop operation, that results in the greatest number of fuel rods subject to boiling transition, and describe the consequences of this event assuming that control rods do not insert (no credit taken for redundant or diverse SCRAM signals or control rod insertion devices).

NMPNS Response 10b

As Anticipated Transients without Scram are beyond design basis events and involve more than one failure, boiling transition is not the applicable acceptance criterion. For ATWS, the 10 CFR 50.46 criteria for fuel integrity have been adopted and peak cladding temperatures are calculated to be well below 2200°F. Therefore, boiling transition is not a fuel integrity criterion. An inadvertent two-pump trip would result in a power decrease as flow is reduced to natural circulation. There would be no boiling transition consequences. An automatic scram may not be generated unless the core is unstable. The stability protection hardware would scram the reactor to protect the fuel in these situations.

The subject of ATWS with instability has been covered generically for the BWR fleet in the following topical reports: GE Nuclear Energy, "ATWS Rule Issues Relative to BWR Core Thermal-Hydraulic Stability", NEDO-32047-A, June 1995, and GE Nuclear Energy, "Mitigation of BWR Core Thermal Hydraulic Instabilities in ATWS," NEDO-32164, December 1992 (note: acceptance of NEDO-32164 is contained in the SER for NEDO-32047-A). NEDO-32047-A describes that for ATWS with instability the fuel integrity criterion is that fuel damage be limited so as not to significantly distort the core, impede core cooling, or prevent safe shutdown. The potentially limiting non-isolation ATWS event with respect to fuel integrity has been determined in NEDO-32047-A to be a turbine trip with full bypass capacity. The full bypass capacity is more limiting than when only partial bypass is available because the full bypass capability eliminates the interference that safety relief valve (SRV) cycling will have with the instability oscillations. This event also results in a large FW temperature reduction, which also aggravates the potential instability. NMP2 has a much smaller bypass capacity and is bounded by the generic study. Another event than can lead to instability is a two pump trip. This event would have a similar behavior without as much feedwater temperature decrease. Non-isolation ATWS events do not put a demand on the reactor vessel as there is no pressurization and no energy is transferred to the suppression pool. Therefore, vessel and containment integrity criteria are met.

If one of these limiting non-isolation events occurs with a core instability and without a scram, then emergency operating procedures require operator action to reduce water level to below the feedwater sparger. This reduces the core subcooling, oscillation magnitude and mitigates the effect on fuel cladding heat up to meet the acceptance criteria.

NRC Question 11

Regarding the LOCA:

NRC Question 11a

Provide a table that describes the break spectrum and single failures analyzed to determine the licensing basis PCT.

NMPNS Response 11a

The analysis of record has confirmed the limiting single failure for NMP2 is the HPCS DG. The limiting break location in the recirculation line has similarly been previously determined.

Given that basis, a table of the break sizes and cases analyzed follows:

Table 11a-1
Break Sizes Analyzed for NMP2 MELLLA Analysis Update

BREAK / ANALYSIS TYPE	POWER SHAPE & RESULT	BREAK SIZE
DBA break – Appendix K:		
Rated flow case 100% Power/100% Flow	mid-peaked power shape (PCT: 1419°F)	3.131ft^2
MELLLA low flow case 100% Power/80% Flow	mid-peaked power shape (PCT: 1411°F)	3.131ft^2
Small Break - Appendix K:		
Updated Calc., 6 ADS* 100% Power/100% Flow	mid-peaked power shape (PCT: 1341°F)	0.07 ft^2
Updated Calc., 6 ADS* 100% Power/100% Flow	top-peaked power shape (PCT: 1478°F)	0.07 ft^2
Updated Calc., 6 ADS* 100% Power/100% Flow	top-peaked power shape (PCT: 1468°F)	0.08 ft^2
Updated Calc., 6 ADS* 100% Power/100% Flow	top-peaked power shape (PCT: 1435°F)	0.09 ft^2
Updated Calc., 6 ADS* 100% Power/100% Flow	top-peaked power shape (PCT: 1402°F)	0.06 ft^2
Small Break – Nominal:		
Updated Calc., 6 ADS* 100% Power/100% Flow	top-peaked power shape (PCT: 1154°F)	0.08 ft^2
Updated Calc., 6 ADS* 100% Power/100% Flow	top-peaked power shape (PCT: 1118°F)	0.10 ft^2
Updated Calc., 6 ADS* 100% Power/100% Flow	top-peaked power shape (PCT: 1164°F)	0.09 ft^2
Updated Calc., 6 ADS* 100% Power/100% Flow	top-peaked power shape (PCT: 1101°F)	0.07 ft^2
Small Break - Appendix K:		
MELLLA low flow case, 100% Power/80% Flow	mid-peaked power shape (PCT: 1329°F)	0.08 ft^2
MELLLA low flow case, 100% Power/80% Flow	top-peaked power shape (PCT: 1470°F)	0.08 ft^2
MELLLA low flow case, 100% Power/80% Flow	top-peaked power shape (PCT: 1419°F)	0.09 ft^2
MELLLA low flow case, 100% Power/80% Flow	top-peaked power shape (PCT: 1454°F)	0.07 ft^2

* Six Safety Relief Valves are dedicated to the Automatic Depressurization System (ADS) and credited for action in the analysis.

NRC Question 11b

Provide the results of licensing analyses that demonstrate compliance with all Title 10 of the *Code of Federal Regulations*, Part 50, Section 50.46 acceptance criteria. Namely provide the oxidation and hydrogen generation results and a discussion of the applicability of NEDO-20566A to MELLLA with increased core flow.

NMPNS Response 11b

Oxidation results for the limiting case show a maximum local oxidation of 0.16% and a core wide oxidation of less than 0.02%. This compares favorably with results for the DBA and limiting Small Break cases from the analysis of record and updated rated and MELLLA analysis cases reported in Attachment (7) of the LAR. It can be seen there are no significant changes as a result of the extended operating domain.

NEDE-20566-P-A presents an analysis on hydrogen generation. Taking a cladding sector at the maximum allowed cladding temperature of 2200°F, it is demonstrated that the maximum hydrogen generation which could result would be on the order of 0.41%, well below the 1% Acceptance Criterion. As a result, the conclusion is drawn that demonstrating PCT within its Acceptance Criterion is sufficient to infer compliance with the hydrogen generation Acceptance Criterion as well.

In regards to the general applicability of NEDE-20566-P-A to MELLLA with increased core flow, this comparison of oxidation results substantiates the evaluation of Section 7.0 of Attachment (7) of the LAR that MELLLA has a negligible effect on compliance with the remaining Acceptance Criteria of 10 CFR 50.46. The continued applicability of the ECCS-LOCA models to the ARTS/MELLLA application and suitability to show conformance to Acceptance Criteria is presented in a letter from R. L. Gridley (GE) to D. G. Eisenhut (NRC), "Review of Low Core Flow Effects on LOCA Analysis for Operating BWRs – Revision 2," May 8, 1978. The NRC review of this letter is documented by letter from D. G. Eisenhut (NRC) to R. L. Gridley (GE), "Safety Evaluation Report of Revision of Previously Imposed MAPLHGR (ECCS-LOCA) Restrictions for BWRs at less than Rated Flow," May 19, 1978.

NRC Question 11c

Verify that LOCA analyses are performed with concurrent loss of offsite power. Provide a list of the single failures that were considered as part of the licensing basis.

NMPNS Response 11c

The NMP2 ARTS/MELLLA ECCS-LOCA evaluation was performed with the standard evaluation model procedure, which invokes no beneficial credits for any offsite power source. This has the equivalent effect of performing the analysis with concurrent Loss-of-Offsite Power (LOOP).

For the NMP2 ECCS-LOCA analysis, the bounding failure is determined by considering the following single failure candidates:

Division 1 DG –	removing Low Pressure Core Spray (LPCS) and one train of Low Pressure Core Injection (LPCI)
Division 2 DG –	removing two trains of LPCI
Division 3 DG –	removing High Pressure Core Spray

NRC Question 11d

Section 8.5 of Attachment (7) of your request describes the analysis of the vessel annulus loading and states: "For the feedwater line break, MELLLA implementation will result in a compartment differential pressure increase of less than 2.25 percent. ... For breaks other than the feedwater line, MELLLA implementation will result in an increase in compartment differential pressure of as much as 6.8 percent for full power conditions and as much as 3.0 percent in the vicinity of the minimum flow point on the MELLLA line." Provide a table describing the breaks and associated initial core conditions that were assumed in the analysis as well as the differential pressure change for each case considered. Provide a qualitative discussion to justify that the limiting case has been considered. Describe the limiting break scenario for the current licensed operating domain.

NMPNS Response 11d

The MELLLA analysis specifically models the recirculation discharge line break (RDLB) and the feedwater line break (FWLB) cases. Based on the analysis of record, the RDLB and FWLB cases are the limiting breaks for annulus pressurization loads. For NMP2, the recirculation suction line break (RSLB) case is not a limiting case for annulus pressurization loads, due to the flow diverters installed in the recirculation suction line biological shield wall penetrations.

The sensitivity of peak break compartment pressure to mass and energy release rate changes is used to estimate the increase in break compartment pressure associated with calculated mass and energy release rate increases. [[ ]] however, detailed compartment pressurization analyses were used to confirm this relationship. The impact on loads is based on the [[ ]]

Table 11d-1 documents the vessel and feedwater line conditions used in the MELLLA evaluation together with the conservative estimates of the impact of MELLLA on the peak compartment differential pressures.

Table 11d-1
MELLLA Impact Evaluation Inputs, Assumptions and Results

Power Level	Feedwater Assumption	Power/Flow Map Region	Power/Flow Point	Vessel Dome Pressure	Downcomer Enthalpy	Feedwater Line Pressure	Feedwater Temperature	Estimated Differential Pressure Increase (4)
Rated Conditions	NFWT(1)	ELLLA	102% CLTP / 87% Flow	1055 psia	528.5	1081 psia	427.4	RDLB < 6.80% FWLB < 2.25%
		MELLLA	102% CLTP / 80% Flow	1055 psia	526.4	1081 psia	427.4	
	NFWT(1) - 20°F	ELLLA	102% CLTP / 87% Flow	1055 psia	525.7	1081 psia	407.2	RDLB < 6.80% FWLB < 2.25%
		MELLLA	102% CLTP / 80% Flow	1055 psia	523.3	1081 psia	407.2	
Off-rated Conditions	NFWT(1)	ELLLA	58.14% CLTP / 34% Flow	1001 psia(2)	499.7	1001 psia(3)	368.8	RDLB < 3.00% FWLB(5) < 2.25%
		MELLLA	60.615% CLTP / 34% Flow	1003 psia(2)	498.7	1003 psia(3)	372.8	
	NFWT(1) - 20°F	ELLLA	58.14% CLTP / 34% Flow	1000 psia(2)	496.8	1000 psia(3)	352.8	RDLB < 3.00% FWLB(5) < 2.25%
		MELLLA	60.615% CLTP / 34% Flow	1002 psia(2)	495.7	1002 psia(3)	356.6	

(1) NFWT = Normal Feedwater Temperature at CLTP (425.1°F)
[[

]]

The limiting line breaks for NMP2 annulus pressurization loads are the feedwater line break and the recirculation discharge line break. Both breaks are modeled as instantaneous breaks at the respective nozzle safe ends. The mass and energy release rates for both breaks are generated with the NEDO-24548 (Technical Description Annulus Pressurization Load Adequacy Evaluation, January 1979) instantaneous break methodology.

The ARTS/MELLLA evaluation addresses the impact of MELLLA implementation at both rated and off rated conditions.

Feedwater Line Break

[[

]]

1) Feedwater line pressure and feedwater enthalpy are maximized at full power conditions.

2) For a constant enthalpy, both mass and energy release rates decrease for decreased upstream pressure.

3) The feedwater line enthalpy is in a range where decreases in enthalpy, for a constant upstream pressure, produce relatively small increases in the critical mass flux. Therefore, the net effect of the lower enthalpy is a decrease in both the energy release rate and the rate (lbm/sec) at which flashed steam is generated in the break compartment.

Recirculation Discharge Line Break

[[

]]

For the full power case, the 102% CLTP / 80% rated flow defines the end of the MELLLA line.

[[

]]

NRC Question 11e

For the limiting licensing basis LOCA scenario, provide figures showing the transient MCPR, downcomer water level, collapsed liquid bypass level, system pressures, steam line flow, break flow, automatic depressurization system flow, high pressure core spray flow, low pressure core spray flow, low pressure coolant injection flow, total of all egress flows, total of all injection flows, and PCT.

NMPNS Response 11e

MCPR is calculated to confirm initialization of the power distribution, set to the MCPR limit. It is important in the DBA break, which is affected by early boiling transition; however, the limiting licensing basis LOCA scenario reported for NMP2 is a small break case. For small breaks, the core flow coastdown is slow enough to prevent early boiling transition before core uncovery. The MCPR parameter is not relevant to this limiting basis LOCA scenario.

Figures for the remaining parameters are provided for the Small Break 0.07 ft^2 Rated case that is the base for the limiting PCT.

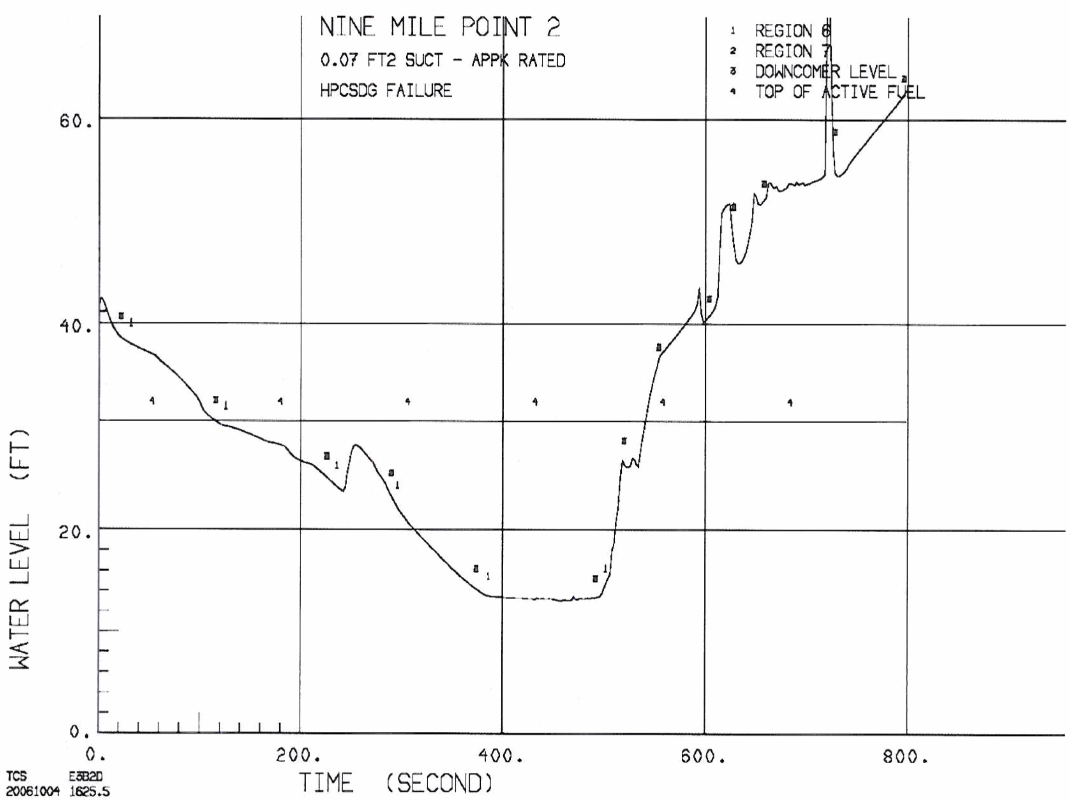


Figure 11e-1 - Downcomer Water Level

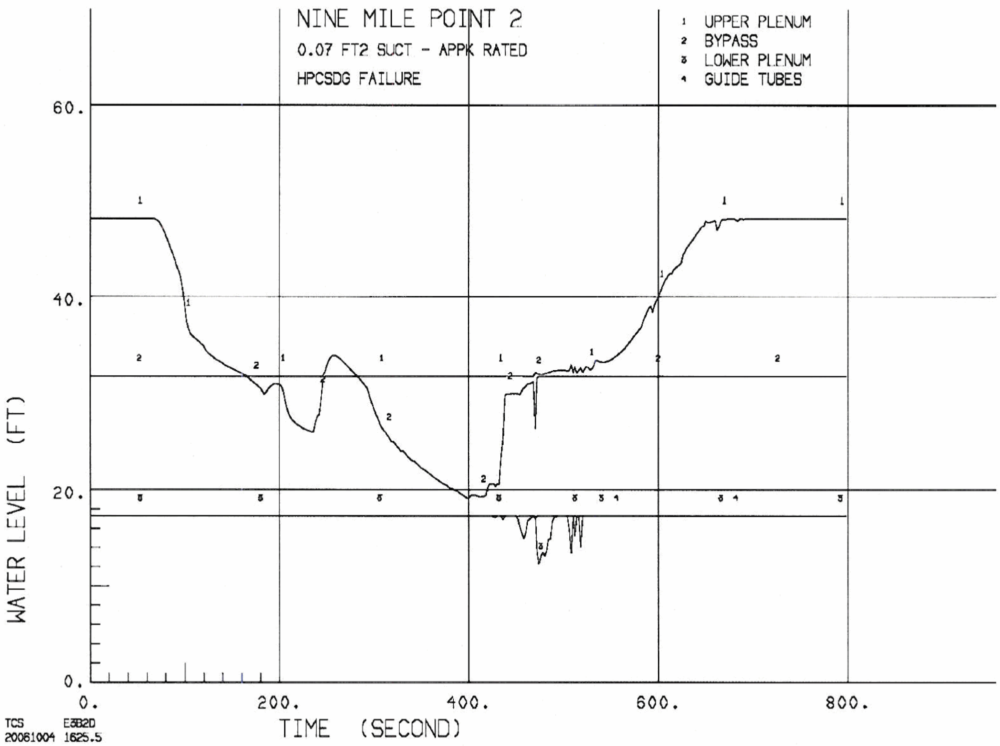


Figure 11e-2 – Water Level for Upper Plenum, Lower Plenum and Bypass Region

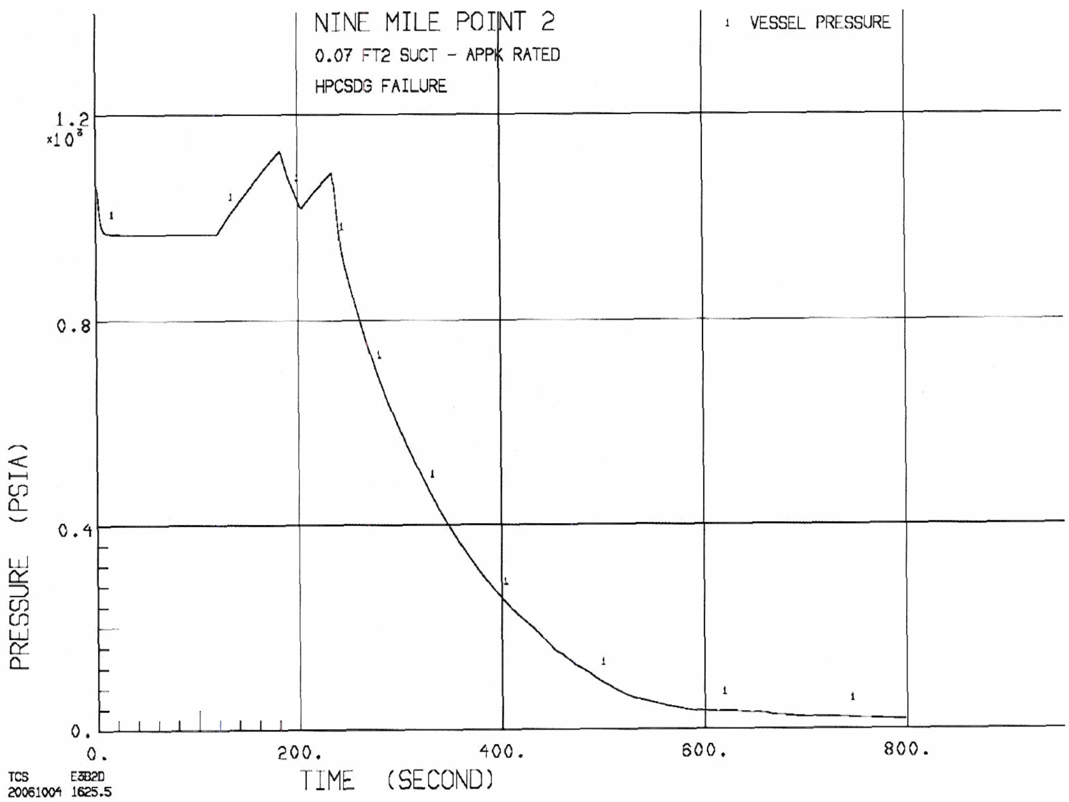


Figure 11e-3 – System (Reactor Vessel) Pressure

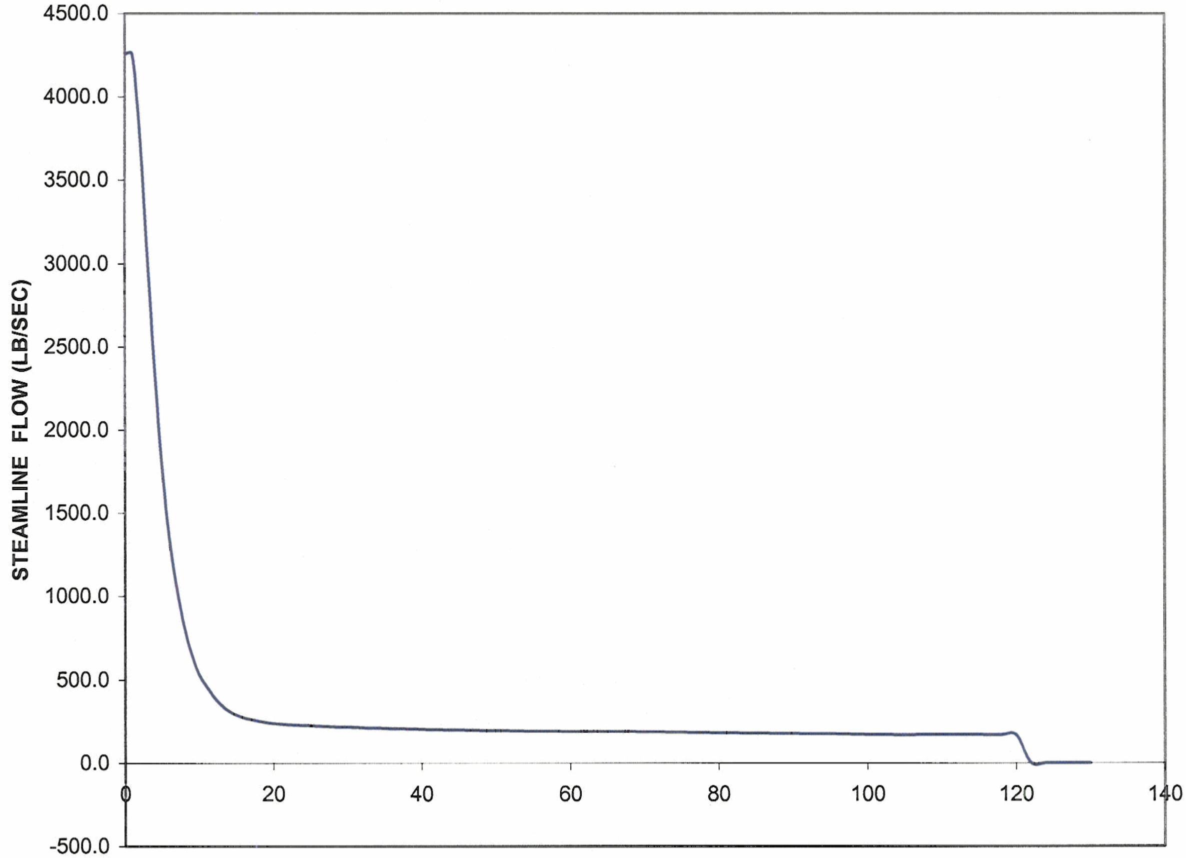


Figure 11e-4 – Steamline Flow

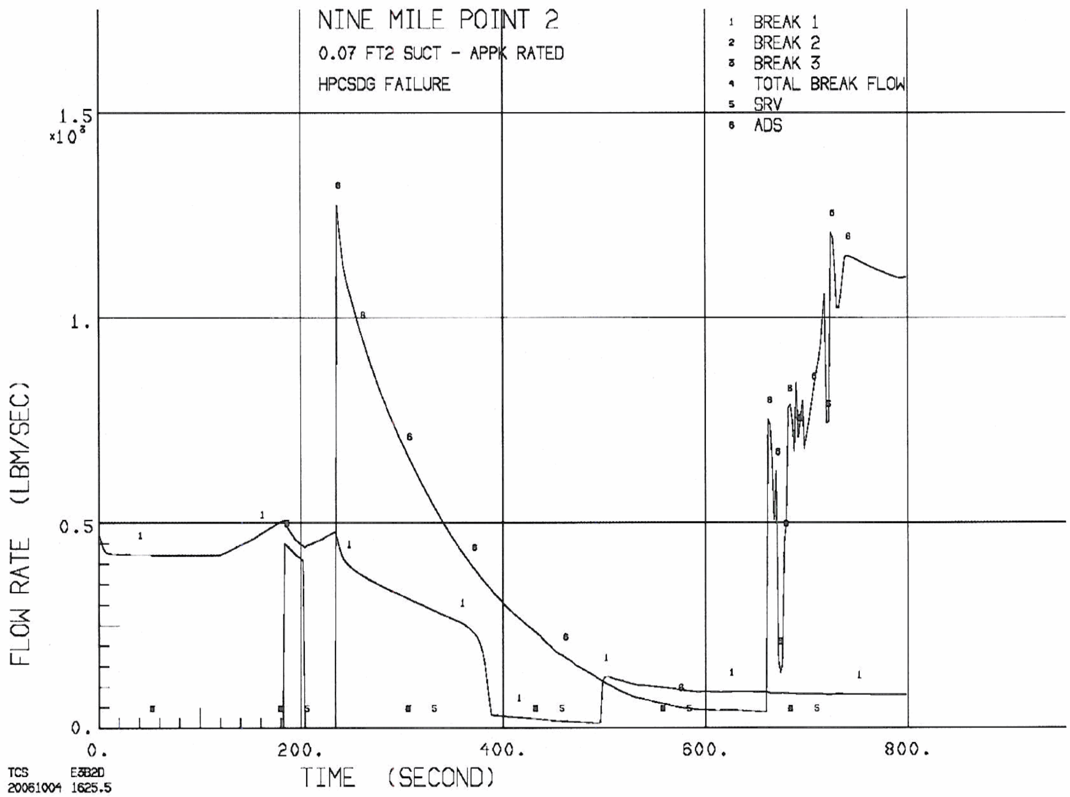


Figure 11e-5 – Break Flow (Egress Flow)

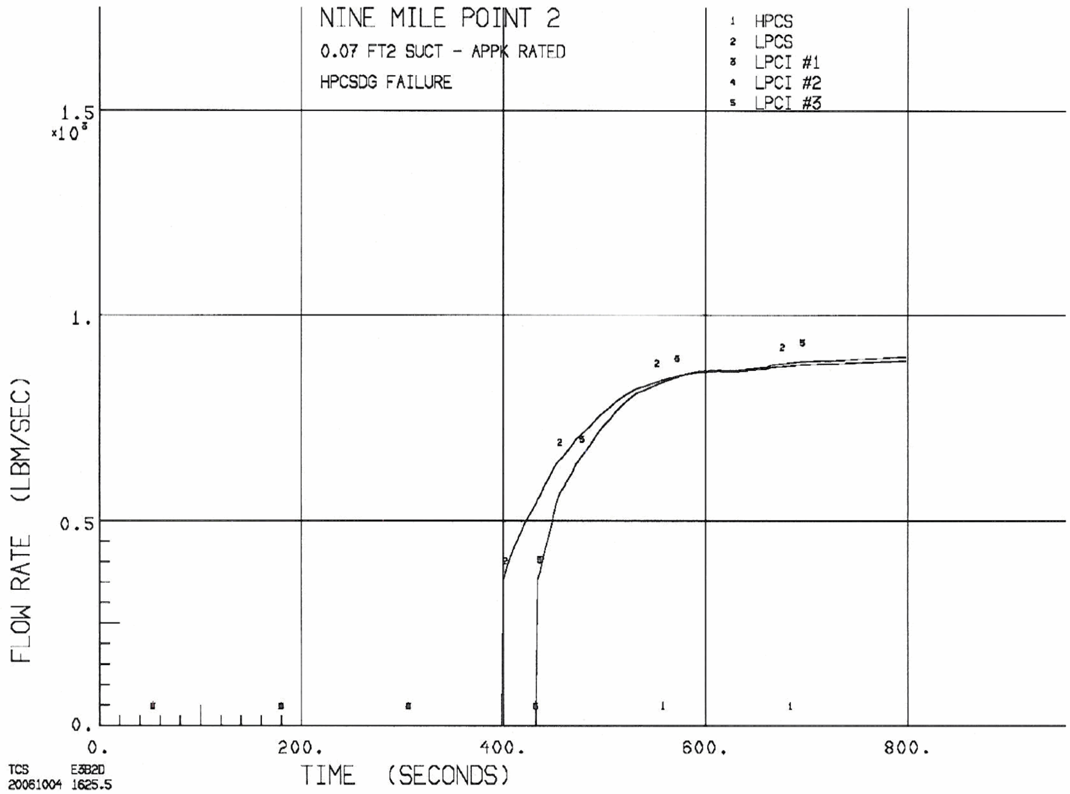


Figure 11e-6 – HPCS, LPCS and LPCI Flow (Injection Flow)

Note: HPCS flow is zero since the assumed single failure is the HPCS DG.

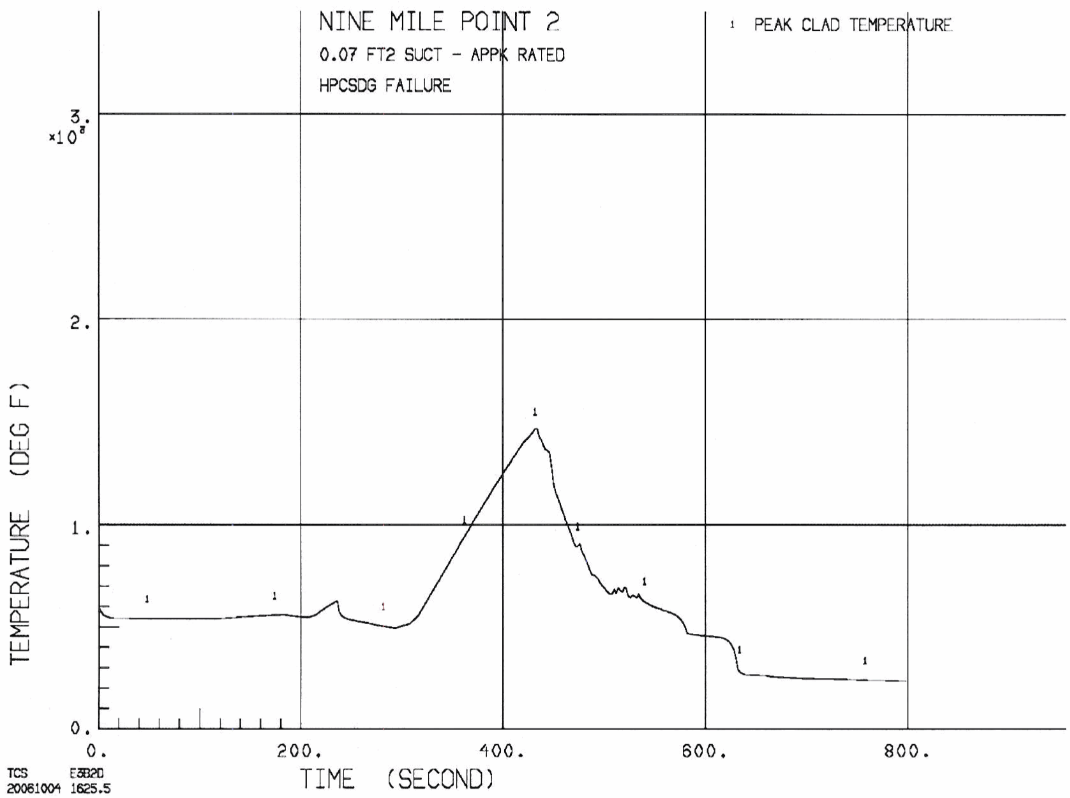


Figure 11e-7 – Peak Cladding Temperature

NRC Question 11f

Evaluate the consequences of a small break occurring as a consequence of a double ended guillotine rupture of the bottom vessel head drain line at MELLLA flow with a top peaked axial power shape. Consider the limiting break size as determined by the break spectrum presented in the topical report. Consider the worst single failure and compare the PCT to the licensing basis PCT.

NMPNS Response 11f

The PCT for the bottom head drain line break exclusively is not calculated. The bottom head drain line is included in analysis of the small break as the evaluation model is applied. The small break area includes the full guillotine bottom head line break area plus additional recirculation suction line area to obtain the total break area represented. With this procedure, the consequences of the double ended guillotine rupture of the bottom vessel head drain line is always covered by the small break spectrum, including consideration of single failure and break location. Therefore, the small break spectrum results for the MELLLA condition with a top peaked power shape presented in Attachment (7) of the LAR already consider the effects of maximum break of the bottom vessel head drain line.

NRC Question 12

Regarding operations:

NRC Question 12a

Provide a figure on the same scale as Figure 1-1 of Attachment (7) of your request that shows and defines the previous and updated average power range monitor (APRM) rod block trip and SCRAM setpoints, the proposed MELLLA operating domain and APRM setpoints, and the Extended Load Line Limit Analysis operating domain and APRM setpoints.

NMPNS Response 12a

A power/flow map with the requested information is provided in Figure 12a-1. At core flows less than Line "C" (Low Recirc Pump Speed, Both Flow Control Valves [FCVs] Max Position) the flow-biased (FB) setpoint curves drop sharply toward a continuation of the Lower Bound Natural Circulation Line ("A") as a result of the correlation of drive flow versus core flow. In the ELLLA operating domain, there is no APRM Rod Block clamp setpoint; as such, the ELLLA APRM Rod Block setpoint line is shown extending to 105% core flow. The ELLLA APRM Flow Biased Scram, and the MELLLA Flow Biased Rod Block and Scram setpoint lines are shown as a combination of the Flow Biased nominal trip setpoints and clamp values. The APRM Flow Biased Scram clamp is not modified for MELLLA.

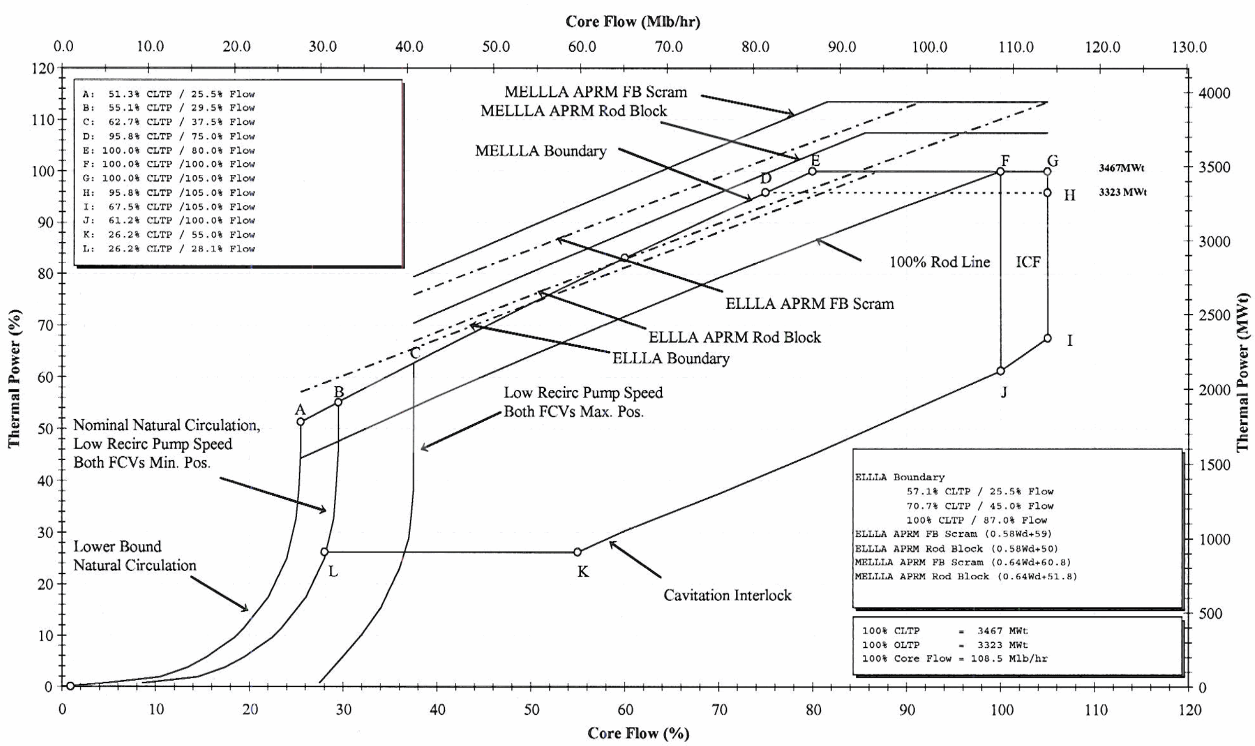


Figure 12a-1

Note that the designators for the curves beginning at points "A" and "B" have been revised from those used in Figure 1-1 of Attachment (7) of the LAR to enhance clarity.

NRC Question 12b

Provide a list and associated description of those updates and upgrades related to the NUMAC-PRNM to support operation in the MELLLA domain.

NMPNS Response 12b

The following NUMAC-PRNM updates are implemented to support operation in the MELLLA domain.

- Two Loop Operation APRM Simulated Thermal Power (STP) Flow Biased Rod Block Setpoint is updated
- Two Loop Operation APRM STP Flow Biased Scram setpoint is updated
- An APRM STP Rod Block Clamp setpoint is added to both Single Loop Operation and Two Loop Operation

NRC Question 13

List all of the SCRAM signals that would be encountered, but not credited, in the determination of the peak vessel pressure following a MSIV closure before crediting the flux SCRAM.

NMPNS Response 13

The only scram signal that would be encountered in a MSIV closure is the MSIV position switch scram. This scram, which is not credited in the analysis, would occur before the MSIV position reaches the analytical limit of 85% open for NMP2, scramming the reactor approximately 2.5 sec before the MSIVs would be full closed at their minimum allowed closure time. If this scram does not occur, the next scram signal is the high neutron flux scram, which occurs about 1.8 seconds into the event due to the closure of the MSIVs.

NRC Question 14

The RBM withdrawal permissive removed trip set points are predicated on a SLMCPR value of 1.07. For single-loop operation (SLO), the SLMCPR is 1.09 for NMP2. Describe how the RBM ensures that the fuel does not exceed SAFDLs during a RWE during SLO.

NMPNS Response 14

The RBM setpoint is not changed when entering SLO conditions. For a given RBM setpoint, there is an associated operating limit MCPR (OLMCPR). Section 4.3.1 of Attachment (7) of the LAR demonstrates that these MCPR limits do not vary significantly over the power/flow map. As described in Section 3.3.6 of Attachment (7) of the LAR, the OLMCPR is adjusted for SLO operating conditions to account for the difference between the SLMCPR for SLO and the SLMCPR for dual loop operation. This adjustment ensures the SLO SLMCPR is protected for all AOOs, including the rod withdrawal error.

NRC Question 15

In Section 8.2 of Attachment (7) of your request, an initial drywell temperature of 105°F was assumed in the design-basis accident LOCA short-term containment pressure/temperature response analysis for MELLLA. If drywell temperature could be lower than 105°F during plant operations, provide assurance that drywell accident pressure will not exceed design pressure.

NMPNS Response 15

As explained in Section 8.0 of Attachment (7) of the LAR, the design basis DBA-LOCA short-term containment pressure and temperature response analysis of NEDE-31944 (Reference 1 of Attachment (7) of the LAR) assumed an initial drywell temperature of 135°F. Thus, the assumed initial drywell temperature of 105°F for the MELLLA analysis is a more conservative assumption relative to peak drywell pressure. The peak containment pressure calculated with this assumed 105°F initial drywell temperature, as reported in Table 8-2, remains well below the design limits.

Table 8-2 of Attachment (7) of the LAR provides a comparison of the containment response for the rated case with a change in initial drywell temperature. As the footnote states, the "Current", "Current†", and "Rated‡" cases all assume an initial drywell temperature of 135°F. The "Rated", "ICF", "MELLLA", and "MELLLA-MPS" cases are all performed with an assumed initial drywell temperature of 105°F. To provide a comparison on a consistent basis, the reported peak drywell pressure for the "Rated‡" case (135°F) is compared to the results of the "Rated" case (105°F). As described in Section 8, these two cases were run for the same 40-second duration time. This comparison shows an increase of approximately 2 psi in the 40-second maximum drywell pressure as a result of the 30°F lower initial drywell temperature. Thus, if the "Current" analysis (135°F) were to be reanalyzed with a lower initial drywell temperature of 105°F (a decrease of 30°F in assumed initial drywell temperature), a similar increase would be anticipated for this case with a peak pressure of approximately 39 psig, which would still be significantly below the design limit of 45 psig for the containment.

While the 105°F initial drywell temperature is a conservative assumption, it is conceivable that the temperature could possibly be lower (e.g., during startup). As discussed above, the decrease of 30°F in assumed initial drywell temperature shows an increase of only about 2 psi in peak drywell pressure. Based on a qualitative assessment, it is concluded that, while further reductions in initial drywell temperature would result in additional increases in long-term peak drywell pressure, they would remain below the design limit.

Thus, it is concluded that even if drywell temperature is lower than 105°F during plant operations, drywell accident pressure will not exceed design pressure.

NRC Question 16

In Tables 8-1 and 8-2 of Attachment (7) of your request, there is a nomenclature difference between Case No. 4 ("Low Pump Speed MFCV-MELLLA") in Table 8-1 and the last case ("MELLLA-MPS") in Table 8-2. Please confirm that they are one and same.

NMPNS Response 16

The "Low Pump Speed MFCV-MELLLA" case of Table 8-1 and the "MELLLA-MPS" case of Table 8-2 are one and the same case.

ATTACHMENT (2)

GEH AFFIDAVIT

GE-Hitachi Nuclear Energy Americas LLC

AFFIDAVIT

I, Richard E. Kingston, state as follows:

(1) I am Vice President, Methods Licensing, Regulatory Affairs, GE-Hitachi Nuclear Energy Americas LLC ("GEH"), 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 1 of GEH's letter, GE-PPO-JXAAQ-KG1-100, M. James to Gary Pavis, entitled ARTS/MELLLA - GEH Responses to RAIs 1, 2, 3, 4, 6, 7, 8, 10b, 11, 13, 14, 15, and 16, October 12, 2007. GEH proprietary information in Enclosure 1, which is entitled "GEH Responses to RAIs 1, 2, 3, 4, 6, 7, 8, 10b, 11, 13, 14, 15, and 16", is identified by a dotted underline inside double square brackets [[This sentence is an example.{3}]].. 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 or licensee, GEH 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 GEH's competitors without license from GEH 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 GEH customer-funded development plans and programs, resulting in potential products to GEH;

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 GEH, 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 GEH, 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, or subject to the terms under which it was licensed to GEH. Access to such documents within GEH 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 for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GEH 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 of analytical model, methods and processes 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 GEH's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GEH'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 GEH.

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.

GEH's competitive advantage will be lost if its competitors are able to use the results of the GEH 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 GEH 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 GEH of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing and obtaining 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 11th day of October, 2007.

Richard E. Kingston

Richard E. Kingston
GE-Hitachi Nuclear Energy Americas LLC