February 27, 2008

Mr. Keith J. Polson
Vice President Nine Mile Point
Nine Mile Point Nuclear Station, LLC
P. O. Box 63
Lycoming, NY 13093

SUBJECT: NINE MILE POINT NUCLEAR STATION, UNIT NO. 2 - ISSUANCE OF AMENDMENT RE: IMPLEMENTATION OF ARTS/MELLLA (TAC NO. MD5233)

Dear Mr. Polson:

The Commission has issued the enclosed Amendment No. 123 to Renewed Facility Operating License No. NPF-69 for the Nine Mile Point Nuclear Station, Unit No. 2 (NMP-2). The amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated March 30, 2007, as supplemented by letters dated October 16, 2007, and November 2, 2007.

The amendment changes the NMP2 TSs to reflect an expanded operating domain resulting from implementation of Average Power Range Monitor/Rod Block Monitor/Technical Specifications/Maximum Extended Load Line Limit Analysis (ARTS/MELLLA).

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly *Federal Register* notice.

Sincerely,

/RA/

Marshall J. David, Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-410

Enclosures:
1. Amendment No. 123 to NPF-69
2. Safety Evaluation

cc w/encls: See next page

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<u>DISTRIBUTION:</u>
(See next page)

Package No.: ML080230243
Amendment No.: ML080230230
Tech Spec No.: ML NRR-058

OFFICE	LPLI-1/PM	LPLI-1/LA	EEEB/BC*	EICB/BC*	SCVB/BC*	SRXB/BC*	SNPB/BC*	ITSB/BC(A)	OGC	LPLI-1/BC
NAME	MDavid	SLittle	GWilson	WKemper	RDennig	GCranston	AMendiola	GWaig	LSubin	MKowal
DATE	1/23/08	1/23/08	12/11/07	12/19/07	12/21/07	12/07/07	12/07/07	2/04/08	2/11/08	2/26/08

*SE transmitted by memo on date shown.
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DATED: February 27, 2008

AMENDMENT NO.   TO RENEWED FACILITY OPERATING LICENSE NO. NPF-69 NINE MILE POINT, UNIT NO. 2

PUBLIC	
LPLI-1	
SLittle	RidsNrrLASLittle
MDavid	RidsNrrPMMDavid
OGC	RidsOGCRp
GHill (2)	
GWilson	RidsNrrDeEeeb
NPatel	
WKemper	RidsNrrDeEicb
BMarcus	
RDennig	RidsNrrDssScvb
RLobel	
RKaripineni	
GCranston	RidsNrrDssSrxb
MRazzaque	
AMendiola	RidsNrrDssSnpb
PYarsky	
TKobetz	RidsNrrDirsItsb
ACRS	RidsNrrAcrsAcnwMailCenter
GDentel, RI	RidsRgn1MailCenter

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NINE MILE POINT NUCLEAR STATION, LLC (NMPNS)

DOCKET NO. 50-410

NINE MILE POINT NUCLEAR STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 123
Renewed License No. NPF-69

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Nine Mile Point Nuclear Station, LLC (the licensee) dated March 30, 2007, as supplemented by letters dated October 16, 2007, and November 2, 2007, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-69 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 123 , are hereby incorporated into this license. Nine Mile Point Nuclear Station, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Mark G. Kowal, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the License and Technical Specifications

Date of Issuance: February 27, 2008

ATTACHMENT TO LICENSE AMENDMENT NO. 123

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-69

DOCKET NO. 50-410

Replace the following page of the Renewed Facility Operating License with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove Page	Insert Page
4	4

Replace the following pages of Appendix A, Technical Specifications, with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
i	i
1.1-4	1.1-4
3.1.7-3	3.1.7-3
3.2.4-1	---
3.2.4-2	---
3.3.1.1-4	3.3.1.1-4
3.3.1.1-8	3.3.1.1-8
3.3.2.1-4	3.3.2.1-4
3.3.2.1-6	3.3.2.1-6
3.4.1-1	3.4.1-1
5.6-3	5.6-3

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 123 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-69

NINE MILE POINT NUCLEAR STATION, LLC

NINE MILE POINT NUCLEAR STATION, UNIT NO. 2

DOCKET NO. 50-410

1.0 INTRODUCTION

By letter dated March 30, 2007 (Agencywide Documents Access and Management Systems (ADAMS) Accession No. ML070950196), as supplemented by letters dated October 16, 2007 (ADAMS Accession No. ML072900438), and November 2, 2007 (ADAMS Accession No. ML073090419), Nine Mile Point Nuclear Station, LLC (the licensee) submitted a license amendment request (LAR) for changes to the Nine Mile Point Nuclear Station Unit No. 2 (NMP2) Technical Specifications (TSs). The requested changes would change the NMP2 TSs to reflect an expanded operating domain resulting from 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 allowable value (AV) would be revised to permit operation in the MELLLA region. The current flow-biased rod block monitor (RBM) would be replaced by a power-dependent RBM, which also would require new AVs. The flow-biased APRM simulated thermal power setdown requirement would be replaced by more direct power and flow dependent thermal limits administration. The Surveillance Requirement (SR) for the standby liquid control (SLC) system would be revised to require each SLC pump to deliver required flow at a discharge pressure ≥1325 psig in lieu of ≥1320 psig. In addition, the SLC relief valve setpoint would be increased from 1394 psig to 1400 psig. The proposed TS changes would revise SRs and the Limiting Condition for Operation (LCO) actions and completion times for each applicable operating condition. Finally, the proposed amendment employs a new model for performing the anticipated transients without scram (ATWS) analysis for ARTS/MELLLA conditions. The requested changes would allow additional startup and operating flexibility.

The supplements dated October 16, 2007, and November 2, 2007, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff's initial proposed no significant hazards consideration determination noticed in the *Federal Register* on May 22, 2007 (72 FR 28721).

2.0 REGULATORY EVALUATION

The regulatory requirements and guidance that the NRC staff considered in its review of the application include the following:

- Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36, "Technical specifications," which provides the regulatory requirements for the content required in a licensee's TS. 10 CFR 50.36 requires that the TS include SRs to assure that LCOs will be met. 10 CFR 50.36 states, in part, that where a limiting safety system setting (LSSS) is specified for a variable on which a safety limit (SL) has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded.

- 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," which sets forth acceptance criteria for the performance of the emergency core cooling system (ECCS) following postulated loss-of-coolant accidents (LOCAs).

- 10 CFR 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants," which requires licensees to establish a program for qualifying electrical equipment important to safety. One aspect of this regulation is the requirement to predict the environmental conditions to which this equipment is exposed.

- 10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants," which, in part, specifies the equivalent flow rate, level of Boron concentration, and Boron-10 isotope enrichment required for a boiling-water reactor (BWR) standby liquid control system (SLCS).

- 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 4, "Environmental and dynamic effects design bases," which requires that structures, systems, and components important to safety be designed to accommodate the effects of environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including the LOCA. These environmental effects include those from the discharge of fluids.

- 10 CFR Part 50, Appendix A, GDC 10, "Reactor design," which requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs).

- 10 CFR Part 50, Appendix A, GDC 12, "Suppression of reactor power oscillations," which requires that the reactor core and associated coolant, control, and protection systems be designed to assure that power oscillations that can result in conditions exceeding SAFDLs are not possible or can be reliably and readily detected and suppressed.

- 10 CFR Part 50, Appendix A, GDC 20, “Protection system functions,” which requires that the protection system be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

- 10 CFR Part 50, Appendix A, GDC 22, “Protection system independence,” which requires that the protection system be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in the loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. It also requires that design techniques, such as functional diversity or diversity in component design and principles of operation, be used to the extent practical to prevent loss of the protection function.

- 10 CFR Part 50, Appendix A, GDC 25, “Protection system requirements for reactivity control malfunctions,” requires that the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

- 10 CFR Part 50, Appendix A, GDC 50, “Containment design basis,” which requires that the containment be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from a LOCA.

- 10 CFR Part 50, Appendix K, “ECCS Evaluation Models,” which describes required and acceptable features of the evaluation models used to calculate ECCS performance.

- Regulatory Guide (RG) 1.105, "Setpoints for Safety-Related Instrumentation,” which describes a method acceptable to the NRC staff for complying with the NRC regulations for ensuring that setpoints for safety-related instrumentation are initially within and remain within the TS limits.

- Regulatory Issue Summary (RIS) 2006-17, “NRC Staff Position on the Requirements of 10 CFR 50.36 regarding Limiting Safety System Settings during Periodic Testing and Calibration of Instrument Channels,” dated August 24, 2006.

3.0 TECHNICAL EVALUATION

3.1 Background - BWR Systems

NMP2 is a BWR/5-series reactor, and the current licensed thermal power (CLTP) is 3467 megawatts-thermal (MWt). The operational flexibility of a BWR during power ascension from the low-power, low-flow core condition to the rated high-power, high-flow core condition is restricted by several factors. Also, once rated thermal power (RTP) is achieved, periodic adjustments to core flow and control rod positions must be made to compensate for the reactivity changes due to Xenon buildup and decay, with fuel and burnable poison burnup.

Factors currently restricting plant flexibility at NMP2 in efficiently achieving and maintaining RTP include:

- The current operating power/flow map
- The APRM flow-biased simulated thermal power setdown requirement
- The RBM flow-biased rod block trip

In the LAR and its supplements, the licensee proposed TS changes to address the above restrictions. To support these proposed changes, the licensee provided in the LAR a NMP2 plant-specific ARTS/MELLLA safety analyses, NEDC-33286P, Rev.0 [Ref.1], prepared by the nuclear steam supply system vendor, General Electric Energy Nuclear (GE). The fuel dependent portions of the safety analyses are based on the Cycle 11 core design using GE14 and GE11 fuel. For the fuel dependent portions of the safety analyses, the licensee performed plant and fuel specific analyses to justify operation in the ARTS/MELLLA condition. In general, the limiting AOO minimum critical power ratio (MCPR) calculation and the reactor vessel overpressure protection analysis are fuel dependent. The non-fuel dependent evaluations, such as containment response, are based on the current plant design and configuration.

The function of the licensed allowable power/flow operating map is to define the normal operating condition of the reactor core used in determining the operating safety limits. NMP2 currently operates in the extended load line limit analysis (ELLLA) region up to approximately 108% rod line based on the CLTP and increased core flow (ICF) region up to 105% core flow, which results in a core flow window of 87% to 105% at RTP. The proposed TS change reflects operation of NMP2 in a region that is above the current rated rod line. This extended operating domain is called the maximum extended load line limit (MELLL). The analyses for the specific operating limits associated with the MELLL region, referred to as MELLLA, are performed as part of the standard cycle-specific reload analysis. Implementation of ARTS/MELLLA would allow for more efficient and reliable power ascensions and would allow RTP to be maintained over a wider core flow range, thereby reducing the frequency of control rod manipulations that require power maneuvers to implement.

The function of the RBM is to prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high-power level operation. It does this by blocking control rod movement that could result in violating a thermal limit (the safety limit MCPR (SLMCPR) or the 1% cladding plastic strain limit) in the event of a rod withdrawal error (RWE) event.

The functions of the APRM include:

- Generation of a trip signal to scram the reactor during core-wide neutron flux transients before exceeding the safety analysis design basis
- Blocking control rod withdrawal whenever operation exceeds set limits in the operating map, prior to approaching the scram level
- Providing an indication of the core average power level in the power range

The flow-biased rod block setdown and APRM flow-biased flux scram trip and alarm functions are provided to achieve these requirements.

The proposed implementation of the ARTS/MELLLA improvements will increase the plant operating efficiency by updating the thermal limits requirements to be consistent with current GE methodology and with improvements in plant instrumentation accuracy.

The ARTS improvements include changes to the current APRM system, which require the TS changes described in Section 3.12 of this safety evaluation (SE). The APRM flow-biased simulated thermal power AV varies as a function of reactor recirculation loop flow, but it is clamped such that it is always less than the APRM neutron flux-high AV. The flow-biased RBM AVs will be replaced by power-dependent AVs. The RBM is designed to prohibit erroneous withdrawal of a control rod during operation at high power levels. This prevents local fuel damage during a single RWE. LCO 3.2.4 currently requires the APRM flow-biased simulated thermal power AV to be reduced when the fraction of rated thermal power is less than the maximum fraction of limiting power density (MFLPD). The setdown requirement ensures that margins to the fuel cladding safety limit are preserved during operation at other than rated conditions. As an alternative to adjusting the APRM flow-biased simulated thermal power AV, the APRM gains may be adjusted such that the APRM readings are greater than or equal to 100% times MFLPD. The NMP2 normal operating practice is to adjust APRM gains when required to meet LCO 3.2.4. Each APRM channel is typically bypassed while the required gain adjustment is made. The setdown requirement originated from the Hench-Levy minimum critical heat flux ratio (MCHFR) thermal limit criterion [Ref. 2]. Improved methodologies have subsequently been developed to provide more effective alternatives to the setdown requirement. An update to the thermal limits requirements, which decreases the dependence on the local thermal hydraulic conditions including the core peaking factors, was developed by GE. The resulting General Electric Thermal Analysis Basis critical power ratio (CPR) correlation model [Ref. 3], which relies on bundle boiling length and exit quality, has been reviewed and approved previously by the NRC staff.

As part of the implementation of ARTS/MELLLA, the flow-biased simulated thermal power AV setdown requirement would be replaced by more direct power and flow dependent thermal limits to reduce the need for manual setpoint adjustments and allow more direct thermal limits administration. Although it is part of the current NMP2 design configuration and TSs, the APRM flow-biased simulated thermal power AV is not credited in any specific NMP2 safety analysis. The proposed AV change would permit operation in the MELLLA region for operational flexibility purposes.

3.2 Background - Instrumentation and Controls Systems

NMP 2 currently uses the digital nuclear measurement analysis and control (NUMAC) power range neutron monitoring system (PRNMS). As part of ARTS/MELLLA implementation, the current flow-biased RBM would be replaced by a power-dependent RBM. The change to a power-dependent RBM can be accomplished with the current NUMAC PRNMS hardware. The change from the flow-biased RBM to a power-dependent RBM would require new AVs. Also, the change to a power-dependent RBM would eliminate the need to maintain flow-dependent RBM – Upscale AVs for two loop and single recirculation loop operation. This would allow removal of the LCO 3.4.1 restriction to reset the RBM – Upscale AV when entering single loop operation.

The APRM flow-biased scram and rod block trip setpoints would be revised to permit operation in the MELLLA region. The flow-biased APRM scram and rod block trip setdown requirement would be replaced by more direct power and flow dependent thermal limits to reduce the need for manual setpoint adjustments and to allow more direct thermal limits administration during operation. Operation in the MELLLA region would provide improved power ascension capability by extending plant operation at RTP with less than rated core flow and result in the need for fewer control rod manipulations to maintain RTP during the fuel cycle.

By SE dated September 5, 1995, the NRC staff-approved GE Licensing Topical Report (LTR) NEDC-32410P, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function." This LTR addressed the full scope of the modification to replace the power range monitoring portion of an analog neutron monitoring system in GE BWRs with a GE NUMAC PRNM including an oscillation power range monitor (OPRM). In this LTR, the NRC staff approved proposed TS changes for APRM reactor trip and rod-block protective functions.

By SE dated August 15, 1997, the NRC staff-approved Supplement 1 to NEDC-32410P (herein, both referred to as NUMAC PRNM LTR), which includes TS requirements for an OPRM and clarifies issues related to the APRM. The NUMAC PRNM LTR describes, in detail, the generic NUMAC PRNM design and several plant-specific variations and plant-specific actions.

3.3 Method of Analysis

The analyses that were used to justify operation with the ARTS improvement and the MELLLA power/flow operating map for a core design using GE14 and GE11 fuels are based on the NSSS vendor (GE) computer codes, methodologies, and applicable industry standards, which are discussed in the LAR, including NEDC-33286P, and in the supplements to the LAR. Table 1-1 of NEDC-33286P lists the GE computer codes used in the safety analyses.

The analyses performed are based on the current plant operating parameters. For the transient and stability tasks (NOTE: NEDC-33286P refers to the different ARTS/MELLLA analyses as tasks), the NMP2 Cycle 11 core design was utilized. NEDC-33286P states that these tasks will be revalidated as part of the subsequent cycle-specific reload licensing analyses in accordance with GESTAR II [Ref. 4]. The NRC staff finds the licensee's method of analysis for the NMP2 MELLLA operation acceptable.

3.4 Fuel Thermal Limits

The potentially limiting AOOs and accident analyses were evaluated to support NMP2 operation in the MELLLA region with ARTS off-rated limits. The power/flow state points chosen for the review of AOOs include the MELLLA region and the current licensed operating domain for NMP2.

The core-wide AOOs included in the current Cycle 11 reload licensing analyses [Ref. 5] and the NMP2 Updated Safety Analysis Report (USAR) [Ref. 6] were examined for operation in the MELLLA region (including off-rated power and flow conditions). The following events were considered by the licensee as potentially limiting in the MELLLA region and were reviewed by the licensee as part of the ARTS program development:

- Generator load rejection with no bypass (LRNBP) event
- Turbine trip with no bypass (TTNBP) event
- Feedwater controller failure (FWCF) maximum demand event
- Loss of feedwater heating (LFWH) event
- Fuel loading error (FLE) event
- Idle recirculation loop start-up (IRLS) event
- Recirculation flow increase (RFI) event
- RWE event

The initial ARTS/MELLLA assessment of these events for all BWR type plants concluded that, for plant-specific applications, only the TTNBP, LRNBP, and FWCF events need to be evaluated at both rated and off-rated power and flow conditions. NEDC-33286P states that the analytical methods and input assumptions used for the NMP2 evaluations were consistent with the bases used in GESTAR II. The LFWH, FLE, IRLS, and RFI events were not re-evaluated for the reasons listed below. The RWE event is discussed separately in the next section of this SE.

- The LFWH event was analyzed at 87% and 80% RCF and found to shown significantly greater margin to boiling transition than the TTNBP, LRNBP, and FWCF events. At lower powers, the LFWH event becomes less severe because there is less feedwater to be affected by the loss of a feedwater heater. In its October 16, 2007, response to an NRC staff RAI, the licensee provided an analysis of the LFWH event at 105% RCF. The results showed that the pressurization transient remains bounding. The LFWH event is analyzed on a cycle-specific basis.

- The FLE event is most limiting at maximum power. Therefore, this event was also not considered in the determination of the off-rated limits.

- The IRLS and RFI events are less limiting than the fast pressurization events (TTNBP, LRNBP, or FWCF). As previously stated, these events were considered generically in the development of the ARTS flow-dependent limits, which are generated based on a conservative two pump flow run-up analysis.

The generic assessment of AOOs for ARTS/MELLLA considered the inadvertent actuation of the high-pressure coolant injection system (HPCI). NMP2 is a BWR/5 without an HPCI, but it relies on high pressure coolant makeup from a high pressure core spray (HPCS). The NMP2 USAR and the licensee's October 16, 2007, response to an NRC staff RAI provided the following information regarding an inadvertent HPCS initiation AOO:

- The injection of the cold water through the HPCS spargers above the active core region quenches steam in the upper plenum. The steam flow rates, even at MELLLA conditions, are sufficient to carry the liquid HPCS flow into the steam separators and, subsequently, the remaining liquid is returned to the vessel annulus.

- The pressure regulator responds to the reduction in steam flow and the hydraulic control system closes turbine control valves to maintain turbine impulse pressure. The lower steam flow rates through the main steam line result in a lower steam line pressure drop, and ultimately the vessel dome pressure is reduced.

- The feedwater control system responds to the increase in vessel downcomer level from the increased liquid flow from the separator and reduces the feedwater flow to maintain the vessel inventory.

- The event results in a lower vessel dome pressure at the nominal downcomer level. The HPCS flow is sufficiently small when compared with the core outlet steam flow at nominal and MELLLA conditions to preclude countercurrent flow, hence the introduction of liquid water into the core region. Ultimately the plant stabilizes at a lower vessel pressure, reduced power level, and nominal vessel level. The reduction in power and pressure ensure that the event remains non-limiting at both nominal and MELLLA conditions.

The licensee performed transient analyses at a variety of power and flow conditions during the original development of the ARTS improvement program. These evaluations are applicable for operation in the MELLLA region. The analyses were utilized to study the trend of transient severity without the APRM trip setdown. A database was established by analyzing limiting transients over a range of power and flow conditions. The database includes evaluations that are representative of a variety of plant configurations and parameters such that the conclusions are applicable to all BWRs. The database was utilized to develop a method of specifying plant operating limits (MCPR and LHGR) such that margins to fuel safety limits are equal to or larger than those applied, currently.

The generic evaluations determined that the power-dependent severity trends must be examined in two power ranges. The first power range is between RTP and the power level (P_{bypass}) where reactor scram on turbine stop valve closure or turbine control valve fast closure is bypassed. The analytical value of P_{bypass} for NMP2 is 30% of RTP. The second power range is between P_{bypass} and 25% of RTP. No thermal monitoring is required below 25% of RTP.

The power-dependent MCPR multiplier, K(P), was originally developed for application to all plants in the high power range (i.e., between RTP and P_{bypass}). The values for K(P) increased at lower powers based on the FWCF transient severity trends. As power is reduced from the rated condition in this power range, the generator load rejection with no bypass and turbine trip with no bypass become less severe because the reduced steam flow rate at lower power results in milder reactor pressurization. However, for the FWCF, the power decrease results in greater mismatch between runout and initial feedwater flow, resulting in an increase in reactor subcooling and more severe changes in thermal limits during the event.

Subsequently, it was identified that the turbine control system performance assumptions used when developing the generic power-dependent limits above P_{bypass} did not correspond to the actual turbine control system performance. This issue was documented to the NRC staff in Reference 7. Specifically, the generic power-dependent limits assumed a fast closure of the turbine control valve (TCV) and associated direct scram would occur for all generator load rejections (GLRs) above P_{bypass}. In reality, the power load unbalance (PLU) device will only initiate a fast closure above a certain power level designated as the PLU power level for this report. For powers between the PLU power level and P_{bypass}, a GLR will result in the slow closure of the TCVs causing pressure to increase until the high pressure or high neutron flux scram setpoint is reached, terminating the transient. The PLU power level varies from plant to plant. Therefore, plant-specific power-dependent limits were developed between PLU power level and P_{bypass}. For NMP2, the PLU power level was identified as 48% of RTP.

Between P_{bypass} and 25% power, NMP2 specific evaluations were performed to establish the plant-unique MCPR and LHGR limits in the low power range (below P_{bypass}). These plant-specific limits include sufficient conservatism to remain valid for future NMP2 reloads of GE14 fuel.

Generic flow-dependent MCPR and LHGR limits are applied to NMP2. These generic limits include sufficient conservatism to remain valid for future NMP2 reloads of GE fuel, utilizing the GEXL-PLUS correlation and the GEMINI analysis methods as defined in Reference 4, providing the core flow corresponding to the maximum two recirculation pump runout is $\leq$ 112.0% of rated core flow (RCF).

The rated operating limit minimum critical power ratios (OLMCPRs) and LHGRs are determined by the cycle-specific reload analyses in accordance with Reference 5. At any power/flow state, all applicable off-rated power-dependent (P) and flow-dependent (F) limits are determined: MCPR(P), MCPR(F), LHGR(P), and LHGR(F). The most limiting MCPR (maximum of MCPR(P) and MCPR(F)) and the most limiting LHGR (minimum of LHGR(P) and LHGR(F)) will be the governing limits.

Because the cycle-specific reload fuel analyses will determine the limits for rated and applicable off-rated conditions, and application of the methodology is demonstrated by the analyses performed for the current operating cycle, this approach is acceptable to the NRC staff.

3.5 RWE Analysis

The RWE transient is currently analyzed during the reload fuel licensing analysis for NMP2. The GE RWE methodology, which is currently employed for NMP2, is consistent with GESTAR II. The RWE transient is hypothesized as an inadvertent reactor operator-initiated withdrawal of a single control rod from the core. Withdrawal of a single control rod has the effect of increasing local power and core thermal power, which lowers the MCPR and increases the LHGR in the core limiting fuel rods. The RWE transient is terminated by control rod blocks, which are initiated by the RBM system.

The function of the RBM is to prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high-power level operation. It does this by blocking control rod movement that could result in violating a thermal limit (the 1% plastic strain criterion or the SLMCPR) in the event of the RWE.

The evaluation of the RWE event was performed taking credit for the mitigating effect of the power-dependent RBM. The RBM setpoints were determined based on a statistical analysis. The RBM has three upscale trip levels. The trip levels were determined based on analyses that compare severity of the RWE with different setpoints. The setpoints that were adopted are based on a 95/95 confidence interval assessment that the RWE consequences do not breach the SLMCPR. The analyses were performed assuming conservative LPRM failure assumptions and using NRC staff-approved methods [Ref. 4]. Specific evaluations were performed for the reference NMP2 core to confirm that the maximum linear heat generation rate (MLHGR) limits are met based on the RBM setpoints. On a core-specific basis, it was confirmed that the RBM monitor setpoints adequately ensure cladding integrity protection by comparison to thermal limits.

The NRC staff finds that the statistical evaluation is sufficiently conservative and the analytical results indicate that the implementation of ARTS/MELLLA with the proposed setpoints provides reasonable assurance that the RWE in the MELLLA operating domain will not result in fuel bundles exceeding their SAFDLs.

Based on the analyses provided by the licensee and the fact that NRC staff-approved methodologies were used, the NRC staff concludes that the NMP2 RWE analysis with the digital NUMAC PRNMS and ARTS/MELLLA implementation at CLTP conditions is acceptable.

3.6 Vessel Overpressure

The main steam isolation valve closure with a flux scram (MSIVF) event was used to determine compliance to the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). This event was previously analyzed at the 102% power/105% flow state point for the NMP2 Cycle 11 reload licensing transient analysis. This is a cycle-specific calculation performed in accordance with Reference 5 at 102% of RTP and the maximum licensed core flow (maximum flow is limiting for this transient for NMP2).

Because the implementation of ARTS/MELLLA does not change the maximum core flow, ARTS/MELLLA does not affect the vessel overpressure protection analysis.

3.7 Thermal-Hydraulic Stability

Protection against exceeding SAFDLs as a result of instability events is provided by the OPRM Option III detect and suppress (DSS) long term stability solution (LTS). The OPRM is armed in the range of powers and flow rates where the reactor is susceptible to instabilities. The OPRM period based detection algorithm (PBDA) provides the primary method for detecting instabilities and initiating a reactor scram to suppress them.

The OPRM setpoint is predicated upon a cycle-specific predetermined ΔCPR/ICPR vs. oscillation magnitude (DIVOM). The NRC staff finds that this curve was generated for NMP2 using acceptable methods.

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 pump runback and downshift. The transient was terminated by an OPRM scram. The basis for the OPRM PBDA setpoint is that instabilities naturally initiating from a steady state condition in the low power/flow regions of the power/flow map are suppressed and the OLMCPR is based on a series of predictions of limiting AOOs. In an RAI, the NRC staff requested that the licensee provide analyses to demonstrate that the OPRM setpoint will protect the fuel against boiling transition for an AOO initiated by the same malfunction as the instability on July 24, 2003, considering that the ARTS/MELLLA operating domain encompasses operating points that are potentially more limiting.

In the licensee's October 16, 2007, RAI response, the results of an analysis performed with the TRACG04 code showed considerable margin to boiling transition shortly after the OPRM would have generated a scram signal. Transient plots indicate that the transient MCPR does not fall below 1.15 until approximately 110 seconds into the transient. The OPRM trip is initiated well before this point at approximately 96 seconds. The transient CPR at this time is above 1.4. The analysis provides a demonstration of the conservatism in the OPRM trip point selection to ensure margin to the onset of boiling transition at the time of suppression. Therefore, the NRC staff finds that the DIVOM based setpoint for the OPRM provides sufficient margin to protect the cladding integrity, even under transient conditions initiated from a limiting operating point.

The NRC staff notes that the OPRM armed boundary remains fixed at 60% RCF, which is generically established. In an RAI, the NRC staff requested that the licensee evaluate the acceptability of the armed boundary considering operation along the MELLLA line where the ratio of power to flow at 60% RCF is higher than would be the case along the 100% rod line or ELLLA line. Using the NRC staff-approved ODYSY frequency domain code [Ref. 8], the licensee's October 16, 2007, response provided the results of analyses along the OPRM armed boundary. The analysis along the armed boundary confirms that there is significant margin to core wide and regional instability. Specifically, the calculated channel decay ratio at 60% RCF along the high flow control line was 0.15; values less than 0.56 indicate that regional mode oscillations are highly unlikely. The combination of these analyses provide reasonable assurance at the operation in the MELLLA domain that it does not adversely impact the efficacy of the Option III DSS LTS solution to protect the cladding integrity; and, therefore, the requirements of GDC 12 are acceptably met.

Based on the analyses provided by the licensee and the fact that NRC staff-approved methodologies were used, the NRC staff concludes that the thermal hydraulic stability characteristics of NMP2 with the proposed ARTS/MELLLA implementation at the CLTP conditions are acceptable.

3.8 LOCA Analysis

The ECCS is designed to provide protection against postulated LOCAs caused by ruptures in the primary system piping. The ECCS performance under all LOCA conditions and the analysis models must satisfy the requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K. 10 CFR 50.46 requires that the calculated fuel peak cladding temperature (PCT) following a LOCA not exceed 2200 °F. To ensure that this requirement is met, the maximum average planar LHGR (MAPLHGR) limits are calculated for each fuel cycle. MAPLHGR limits are specified in the cycle-specific Core Operating Limits Report (COLR).

The current licensing basis SAFER/GESTR-LOCA analysis for NMP2 [Reference 9 for GE14 fuel and Reference 10 for GE 11 fuel] was reviewed by the licensee to determine the effect on the ECCS performance resulting from NMP2 operation in the MELLLA domain.

In general, the two major competing phenomena that affect the fuel PCT in the design basis LOCA calculation, which are sensitive to the higher load line in the operating power/flow map, are the time of boiling transition (BT) at the high power node of the limiting fuel assembly and the core recovery time. Initiation of the postulated LOCA at lower core flow may result in earlier BT at the high power node, compared to the 100% of RCF results, causing a higher calculated PCT. On the other hand, initiation of the postulated LOCA at lower core flow (higher power-to-flow ratio, hence higher core inlet subcooling) affects break flow rate, which may result in faster core recovery, compared to the 100% RCF, and can lower the PCT. The net effect on the calculated PCT is acceptable as long as the results remain less than the licensing basis PCT limits.

The nominal and 10 CFR Part 50, Appendix K PCT responses following a large recirculation line break for most plants show that the PCT effect due to MELLLA is small. In some cases, there may be a significant PCT increase if early BT penetrates down to the highest-powered axial node in the fuel bundle. This can happen at core flows in the MELLLA region. For small breaks, the fuel remains in nucleate boiling until uncovery, and MELLLA is expected to have no adverse effect on the small-break LOCA response.

The ARTS-related changes do not affect the LOCA analysis. The current NMP2 licensing basis specifies a requirement in maximum LHGR as a function of drive flow, known as the APRM setdown requirement. This lower LHGR requirement is applicable to core flows lower than 87% of RCF. With the implementation of ARTS, this lower LHGR requirement is being replaced with direct core power and flow fuel thermal limits by the ARTS improvement option. However, these limits are not credited in the LOCA analysis. Therefore, the LOCA analysis is not affected by the implementation of ARTS.

For NMP2, calculations assuming the MELLLA extended operation domain were performed to quantify the effect on PCT to the allowed operation envelope. The MELLLA assumptions for the limiting large recirculation line break case resulted in an insignificant change in the calculated PCT with a resulting PCT that is slightly less than the comparable rated assumption case. Calculations were performed considering a top-peaked power shape applied for small-break sizes. The top-peaked power shape was defined by imposing the peak LHGR and MCPR limits on the core. To assure a limiting case had been identified, a small break spectrum was calculated to complete the analysis. Based on an NRC staff RAI, the licensee submitted the results of the analysis for NRC staff review in its October 16, 2007, response, and the NRC staff verified that the limiting PCT for NMP2 is a small-break LOCA at CLTP/RCF. The licensing basis PCT was determined based on this finding, and it was significantly less than the 2200 °F limit. The LOCA analysis results are shown in the table below.

LOCA Analysis Results

	Licensing Basis Value	10 CFR 50.46 Acceptance Criterion
Peak Clad Temperature	1480 °F	2200 °F
Cladding Oxidation	0.16%	17%
Hydrogen Production	0.41%	1%

MELLLA has a negligible effect on compliance with the other acceptance criteria of 10 CFR 50.46. Because cladding oxidation is primarily determined by PCT, MELLLA can affect the amount of cladding oxidation in those cases where there is a significant PCT increase. Jet pump BWRs have significant margin to the local cladding oxidation and core-wide metal-water reaction acceptance criteria. The compliance with the 2200 °F limit ensures compliance with the local cladding oxidation and core-wide metal-water reaction acceptance criteria for GE14 fuel. Compliance with the coolable geometry and long term cooling acceptance criteria have been demonstrated generically for GE BWRs [Ref. 11]. MELLLA does not affect the basis for these generic dispositions. Therefore, MELLLA has a negligible effect on compliance with the other acceptance criteria of 10 CFR 50.46.

The NMP2 MELLLA evaluation is based on plant-specific calculations with GE14 fuel using SAFER/GESTR methodology [Refs. 11 through 15]. The licensee performed calculations for rated flow and power conditions to establish a baseline PCT, with model changes as have been identified since the last ECCS-LOCA analysis using the SAFER/GESTR methodology [Ref. 9].

The calculations for NMP2 show that the MELLLA option will meet the PCT acceptance criteria for a representative core with GE14 fuel and has no effect on any other LOCA criteria. Therefore, no additional restrictions on fuel power to account for LOCA criteria compliance are required. Calculations at the CLTP/RCF condition result in the highest PCT for both the small and large-break LOCA and set the licensing basis PCT for NMP2. Calculations performed at the CLTP/ MELLLA core flow condition result in lower PCT than the CLTP/RCF condition.

The NRC staff review confirmed that the small-break LOCA spectrum adequately covers an acceptable range in size, single failures, and location to determine the limiting small-break LOCA event for NMP2 and is, therefore, acceptable.

3.9 ATWS

The licensee's ATWS analysis provided in the LAR considered three potentially limiting events: closure of all main steam isolation valves, pressure regulator failure (open) to maximum steam demand, and a loss of offsite power (LOOP). These events were analyzed using the NRC staff-approved ODYN code [Ref. 18] to determine the core integrity using applicable acceptance criteria.

The first two events resulted in reactor pressure vessel (RPV) isolation with a large increase in reactor power due to the pressurization of the RPV. The analyses showed that EOC conditions are most limiting. The NRC staff finds that the analyses were adequately conservative and demonstrated substantial margins to both core and containment acceptance criteria.

The LOOP event was analyzed because it results in the highest RPV pressure after SLCS initiation. The LOOP results in a loss of instrument air resulting in the inability to operate the safety relief valves (SRVs) in the relief mode. The analysis showed that the feedwater and recirculation pump coastdowns reduce reactor power, limiting the consequences of the event.

The ATWS analyses provided by the licensee in the LAR were for isolation events, which result in a pressurization of the RPV, reactor power increase, and maximum suppression pool thermal loads. In an RAI, the NRC staff asked if there is the potential for non-isolation ATWS events to result in more limiting conditions. In its October 16, 2007, RAI response, the licensee provided additional information regarding the consequences of a non-isolation ATWS. In particular, the availability of bypass capability limits vessel pressurization and reduces any load to the suppression pool, thus, these events are non-limiting from a containment and vessel integrity standpoint. The NRC staff agrees with the licensee's determination.

The emergency operating procedures during ATWS events would mitigate the potential for reactor instabilities by requiring that the downcomer level be reduced below the feedwater spargers to decrease the core inlet subcooling. Furthermore, activation of the SLCS would suppress any oscillations by shutting down the reactor. Therefore, the NRC staff finds that the licensee has adequately addressed the consequences from potential ATWS events, selected the most challenging ATWS events in regards to core coolability, containment integrity, and vessel integrity. The licensee has also verified that the potential to damage fuel as a result of instabilities during an ATWS is effectively addressed and mitigated by plant emergency operating procedures. The NRC staff finds this acceptable.

ATWS events are mitigated by the SLCS. The licensee proposed a change in the setpoint pressure for the SLCS relief valve. To ensure compliance with ASME Code, Section III, the licensee has confirmed that the SLCS total accumulated pressure will remain below the maximum pressure of 1540 psig given a relief valve pressure setpoint that is the same as the design pressure of 1400 psig. The NRC staff, therefore, finds that the proposed change is still in compliance with the ASME Code, Section III and is, therefore, acceptable.

The NRC staff finds that the licensee has demonstrated compliance with the requirements of 10 CFR 50.62.

3.10 Containment Systems

NEDC-33286P discusses the methods used and the results obtained for the implementation of ARTS/MELLLA at NMP2. Section 8 of NEDC-33286P discusses the containment systems response.

Extending plant operation into the MELLLA region affects the subcooling of the postulated accident break flow and, therefore, the break flow mass flux as well as the reactor power at states below 100% RTP.

3.10.1 Short-Term Pressure and Temperature Response

The peak drywell pressure, wetwell pressure and maximum drywell-to-wetwell differential pressure occur during the blowdown period of the LOCA.

The licensee analyzed the short-term pressure and temperature response for the following four cases. The licensee stated that these cases cover the full extent of the operation of NMP2 in the MELLLA domain including the ICF condition. Inclusion of ICF provides a more complete range of operating states. The four cases are:

- 102.0% of RTP and 100.0% of core flow (i.e., RCF)
- 102.0% of RTP and 105.0% of core flow (i.e., ICF)
- 102.0% of RTP and 80.0% of core flow (i.e., MELLLA)
- 56.2% of RTP and 29.5% of core flow (Low Pump Speed and Minimum Recirculation Flow Control Valve Position) (MELLLA-MPS)

Table 8-1 of NEDC-33286P provides the corresponding core flow in million pounds per hour, feedwater inlet temperature and reactor vessel dome pressure.

These cases correspond to 2% above the RTP of Points E, F, and G of the power/flow map of Figure 1-1 of NEDC-33286P and Point B of this figure. This selection of points varies the power and the coolant subcooling over the operating range, including ICF. The NRC staff, therefore, considers this selection of cases to be acceptable because the worst case blowdown conditions are included.

The licensee performed the mass and energy release calculations using the LAMB computer code [Ref. 19] and the M3CPT computer code [Ref. 20] to calculate the corresponding containment response. Both computer codes have been used for containment licensing calculations previously found acceptable by the NRC staff.

Table 8-2 of NEDC-33286P provides the results of the short term calculations for the four cases. A portion of this table is repeated below.

	Drywell Pressure (psig)	Wetwell Pressure (psig)	Differential Pressure (psid)	Drywell Temperature (^{0}F)	Wetwell Temperature (^{0}F)
Design Limit	45.0	45.0	25.0	340	270
Current	33.0	26.9	18.1	278	116
Rated	34.9	29.1	18.6	281	116
ICF	34.9	29.2	18.6	281	116
MELLLA	34.8	29.1	18.1	281	115
MELLLA-MPS	34.3	28.6	16.3	280	112

Note: The pressures and temperatures reported in this table represent peak values observed during the first 40 seconds of the event. This time is sufficient for evaluation of the effect of subcooled break flow on these containment parameters. The current values shown in this table are the current licensing basis analysis truncated to the first 40 seconds of the event, and were determined assuming an initial drywell temperature of 135 °F, while the rest of the values shown were determined assuming an initial drywell temperature of 105 °F.

For all cases, the postulated break is the double-ended rupture of a recirculation suction line, which gives the maximum discharge rate into the drywell. Other assumptions were made, which result in maximizing the drywell and wetwell pressure. These are listed in Section 8.2 of NEDC-33286P. The NRC staff agrees that the calculation assumptions maximize the drywell and wetwell pressure.

The NRC staff finds that the calculation results for the four cases are less than the design limits and are, therefore, acceptable.

The licensee stated that the initial containment conditions used in the DBA-LOCA short-term containment pressure/temperature response analysis are identical to those assumed in the current design basis DBA-LOCA short-term containment response analysis [Ref. 21] with the single exception that an initial drywell temperature of 135 °F was assumed in the current analysis whereas an initial drywell temperature of 105 °F is assumed in the MELLLA analysis. The licensee was requested, in an NRC staff RAI, to assure that drywell accident pressure would not exceed design pressure if initial drywell temperature were to be lower than 105 °F.

In its October 16, 2007, response to the NRC staff's RAI, the licensee provided a comparison of the containment response for the "rated" case with a change in initial drywell temperature as shown in Table 8-2 of NEDC-33286P. The comparison showed an increase of approximately 2 psi in the drywell pressure as a result of the 30 °F lower initial drywell temperature. The licensee stated that the 105 °F initial drywell temperature is a conservative assumption, but it is conceivable that the temperature could possibly be lower (e.g., during startup). Based on the margins between the peak calculated pressure of 36.8 psig for the "current" case with a

135 °F initial drywell temperature (not shown in the table above) and the design limit of 45 psig for the containment, the licensee stated that drywell accident pressure will not exceed design pressure even if the drywell temperature happens to be lower than 105 °F during plant operations. The NRC staff concludes that, based on the 8.2 psig margin to the design limit and on the relatively small increase in pressure with a decrease in temperature, the assessment by the licensee provides reasonable assurance that the peak drywell pressure would remain below the design limit.

3.10.2 Hydrodynamic Loads

NEDC-33286P discusses the three major hydrodynamic loads associated with the Mark II containment. These are pool swell, condensation oscillation and chugging loads.

Immediately following the break, water in the downcomers is cleared by the increase in drywell pressure. Following the water clearing, nitrogen from the drywell (NOTE: the NMP2 containment is inerted with nitrogen gas) flows from the drywell through the vent system into the wetwell. The nitrogen bubbles expand forcing the suppression pool level to rapidly increase. This results in impact and drag forces on structures in the wetwell. After passage of the nitrogen through the downcomers, the steam from the drywell is condensed in the suppression pool. The mass flux through the vents decreases with time as the drywell pressure decreases. The condensation rate is nearly steady at high mass flux values, it is then characterized by periodic variations as the mass flux decreases (condensation oscillation). This is followed by intermittent condensation at low mass flux (chugging). Pool swelling, condensation oscillation and chugging result in loads on containment structures. The existing hydrodynamic loads analysis is located in Appendix 6A of the NMP2 USAR.

The licensee stated that the drywell pressure response used in the pool swell design load analysis bounds the initial pressurization predicted for ARTS/MELLLA. Therefore, the NRC staff finds that ARTS/MELLLA operation is acceptable with respect to pool swell loads.

Condensation oscillation loads increase with higher steam mass flux and higher suppression pool temperature. The licensee stated that suppression pool temperature and steam mass flux used to define NMP2 limits bound the values of these parameters calculated with application of ARTS/MELLLA. Therefore, the NRC staff finds that ARTS/MELLLA is acceptable with respect to condensation oscillation.

Chugging occurs at relatively low mass flux values. Steam bubble collapse is random in terms of intensity and time. Low values of steam mass flux occur during a LOCA when ARTS/MELLLA has negligible impact on the containment pressure response. Therefore, the NRC staff finds that the current chugging load definition is not affected by ARTS/MELLLA.

Operation in the ARTS/MELLLA region does not require changes to the safety/relief valve setpoints and, therefore, does not affect the containment loading due to SRV discharge transients.

NEDC-33286P states that the containment hydrodynamic loads analyses for ARTS/MELLLA operation also includes consideration of the currently-licensed 20 °F feedwater heater out-of-service (FWHOOS) and possible future applications for 120 °F final feedwater temperature reduction (FFWTR) and FWHOOS.

3.10.3 Reactor Asymmetric Loads

Section 8.5 of NEDC-33286P addresses the impact on high energy line breaks in the annulus region between the reactor vessel and the shield wall due to the change in the reactor operating domain from the current power/flow map boundary (ELLLA) to the MELLLA power/flow map boundary. The methods used [Ref. 22] are those for the current analysis discussed in the NMP2 USAR.

The licensee stated that, for the feedwater line break, MELLLA implementation results in a differential pressure increase of less than 2.25%. For breaks other than the feedwater line break, MELLLA implementation will result in an increase in compartment differential pressure of up to 6.8% for full power conditions. The increases are due to the slightly higher integrated energy releases that occur for pipe breaks at MELLLA operating conditions. The increases are within available margin and have been determined with acceptable methods and are, therefore, acceptable to the NRC staff.

3.10.4 Long-Term Accident Response

The long-term containment response is not affected by MELLLA operation because the decay heat does not change, and there is negligible difference in the vessel sensible heat in the MELLLA operating domain.

3.10.5 Conclusion

The NRC staff finds that implementation of ARTS/MELLLA is acceptable for NMP2 with respect to containment systems response based on the use of acceptable methods and conservative assumptions. The implementation of ARTS/MELLLA at NMP2 satisfies GDCs 4 and 50 with respect to containment integrity.

3.11 Environmental Qualification of Safety-Related Electrical Equipment

Section 12 of NEDC-33286P states that NMP2 evaluated the effects of the higher mass and energy release profiles resulting from ARTS/MELLLA implementation and concluded that the resulting subcompartment pressures, temperatures and humidity levels are acceptable with respect to the existing design criteria. In an RAI, the NRC staff requested the licensee to provide a detailed explanation to support this conclusion.

In its November 2, 2007, response, the licensee provided the results of detailed evaluations and conservative screening analyses of the steam line break, feedwater line break, and reactor water cleanup system line break. Based on these evaluations and analyses, the licensee determined that:

- The MELLLA operating domain does not change the initial assumed humidity conditions for environmental qualification (EQ) analysis associated with the high energy line break (HELB) condition events. The EQ temperatures are defined based on the conservative assumption of 100% relative humidity and, therefore, are bounding calculations for the HELB events.

- The MELLLA operating domain does not change the operating or accident source term because the maximum steam flow and power level are unchanged; therefore, there is no impact on the radiation qualification envelope.

- The licensee provided information to demonstrate that the relative increases in pressure and/or temperature due to various HELBs were small relative the EQ design envelope margin. The licensee concluded that the current design envelope bounds the effects of ARTS/MELLLA implementation with margin for pressure and temperature.

As a result, the licensee concluded that EQ is maintained with ARTS/MELLLA implementation.

Based on its review of the licensee's evaluations and analyses, the NRC staff finds that EQ will be maintained and that the requirements of 10 CFR 50.49 will be met with ARTS/MELLLA implementation.

3.12 ARTS/MELLLA - Related TS Changes

In the LAR and its supplements, the licensee proposed changes to the NMP2 TS. An evaluation of the changes follows.

3.12.1 TS 3.1.7, Standby Liquid Control (SLC) System

SR 3.1.7.7 currently specifies the following for each SLC pump:

> "Verify each pump develops a flow rate ≥ 41.2 gpm at a discharge pressure ≥ 1320 psig."

The SLC pump discharge pressure would be raised from 1320 psig to 1325 psig. The NRC staff finds this increase acceptable because the revised ATWS analysis resulted in a peak upper plenum pressure that is 5 psi greater than the current analysis, which results in a corresponding 5 psi increase in the required SLC pump discharge pressure.

3.12.2 TS 3.2.4, Average Power Range Monitor (APRM) Gain and Setpoint

This TS, which includes requirements for flow-biased APRM simulated thermal power setdown, would be deleted. The NRC staff agrees that this TS is no longer needed because improved methodologies, with implementation of ARTS/MELLLA, provide more effective alternates to the requirement.

The following additional changes would be made to reflect deletion of TS 3.2.4:

- The TS Table of Contents would be revised.

- The definition for MFLPD would be deleted from TS Section 1.1.

- References to TS Section 3.2.4 would be deleted from SR 3.3.1.1.3.

3.12.3 TS 3.3.1.1, Reactor Protection System (RPS) Instrumentation

The licensee proposed multiple changes to this TS as follows:

- SR 3.3.1.1.3 would be revised to delete gain adjustments required by LCO 3.2.4. Consistent with the proposed deletion of TS 3.2.4, SR 3.3.1.1.3 would be changed from:

"Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power ≤ 2% RTP plus any gain adjustment required by LCO 3.2.4, "Average Power Range Monitor (APRM) Gain and Setpoint," while operating at ≥ 25% RTP."

To:

"Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power ≤ 2% RTP while operating at ≥ 25% RTP."

Because of the deletion of TS 3.2.4, this change is administrative and is, therefore, acceptable to the NRC staff.

- Table 3.3.1.1-1, Function 2.b, Average Power Range Monitors, Flow Biased Simulated Thermal Power - Upscale, would be revised. Specifically, the AV for two-loop operation would be changed from:

  "≤ .58W + 62% RTP and ≤ 115.5% RPT(b)"

  To:

  "≤ .64W + 63.8% RTP and ≤ 115.5% RPT(b)."

  This AV includes the same conservatisms to account for, among others, testing and calibration errors as the original AV value using the GE setpoint methodology described in NEDC-31336 P-A [Ref. 23], with minor adjustments for MELLLA operation made per the methodology in NEDC-33004P-A [Ref. 24]. Both of these LTRs have been approved by the NRC staff as documented in their associated SEs. Therefore, this AV change is acceptable to the NRC staff. The AV for single-loop operation contained in Footnote (b) would not be changed.

3.12.4 TS 3.3.2.1, Control Rod Block Instrumentation

The licensee proposed multiple changes to this TS as follows:

- Table 3.3.2.1-1, Function 1.a, Rod Block Monitor - Upscale, would be revised by replacing the current flow-dependent function with three power-dependent functions. The new functions would be:

  1.a, Rod Block Monitor - Low Power Range – Upscale

  1.b, Rod Block Monitor - Intermediate Power Range – Upscale

  1.c, Rod Block Monitor - High Power Range - Upscale

  The NRC staff finds this acceptable because the proposed TS changes to the RBM functions are consistent with those in the NUMAC PRNM LTR.

- Table 3.3.2.1-1, for new Functions 1.a, 1.b, and 1.c, the current mode of applicability Note (a), “THERMAL POWER ≥ 30% RTP and no peripheral control rod selected,” would be replaced by four notes:

  New Note (a), “APRM Simulated Thermal Power is ≥ 28% and < 63% RTP and MCPR < limit specified in the COLR and no peripheral control rod selected.”

  New Note (b), “APRM Simulated Thermal Power is ≥ 63% and < 83% RTP and MCPR < limit specified in the COLR and no peripheral control rod selected.”

  New Note (c), “APRM Simulated Thermal Power is ≥ 83% and < 90% RTP and MCPR < limit specified in the COLR and no peripheral control rod selected.”

  New Note (d), “APRM Simulated Thermal Power is ≥ 90% RTP and MCPR < limit specified in the COLR and no peripheral control rod selected.”

  New Note (a) would be applicable to Function 1.a, new Note (b) would be applicable to Function 1.b, and new Notes (c) and (d) would be applicable to Function 1.c. The NRC staff finds this acceptable because the proposed notes would be consistent with the calculations in GE document 0000-0053-1006 NMP2 A-M-T506-RBM-Calc-2006, Revision 0, dated January 2007, which is Attachment A to NEDC-33286P, and with the recommendations of the NUMAC PRNM LTR.

- Table 3.3.2.1-1, Function 1.b, Rod Block Monitor- Inop, would be renumbered from 1.b to 1.d; the mode of applicability would be revised from Note (a), “THERMAL POWER ≥ 30% RTP and no peripheral control rod selected,” to Note (d), “APRM Simulated Thermal Power is ≥ 90% and MCPR < limit specified in the COLR and no peripheral control rod selected,” and Note (e), “APRM Simulated Thermal Power is ≥ 28% RTP and < 90% RTP and MCPR < limit specified in the COLR and no peripheral control rod is selected;” and the applicability of SR 3.3.2.1.4 would be deleted. The NRC staff finds this acceptable because the proposed notes and SR are consistent with the recommendations of the NUMAC PRNM LTR.

- Table 3.3.2.1-1, Function 1.c, Downscale, would be deleted. The licensee stated that the RBM Downscale function was used to detect substantial reductions in the RBM local flux after a "null" is completed (NOTE: a null occurs after a new rod selection). This function, in combination with the RBM Inop function, was intended in the original system to detect problems with or abnormal conditions in the RBM equipment and system. Unlike other neutron monitoring system downscale functions (e.g., the APRM Downscale), there are no normal operating conditions that are intended to be detected by the Downscale function. In the licensee’s original analog RBM, the inclusion of the Downscale function, in addition to the Inop function, had merit in that the analog equipment had some failure modes that could result in a reduction of signal, but not a full failure. No credit is taken for the RBM Downscale function in the establishment of the RBM Upscale AVs. The NRC staff concludes that this function can be removed because this was an intended function inherent to the original analog equipment and does not provide added value if included with a digital system.

- Table 3.3.2.1-1, Function 2, Rod Worth Minimizer, current Note (b) would be renumbered as Note (f). With the creation of new Notes (b) through (e), the renumbering of Note (b) as Note (f) is an administrative change and is, therefore, acceptable to the NRC staff.

- Table 3.3.2.1-1, Function 3, Reactor Mode Switch – Shutdown Position, current Note (c) would be renumbered as Note (g). With the creation of new Notes (b) through (e), the renumbering of Note (c) to become Note (g) is an administrative change and is, therefore, acceptable to the NRC staff.

- Table 3.3.2.1-1, Note (h), "Allowable Value specified in the COLR," would be added for Functions 1.a, 1.b, and 1.c. With the deletion of current TS 3.3.2.1, Function 1.c, all AVs would be specified in the COLR, the addition of Note (h) is an administrative change and is, therefore, acceptable to the NRC staff.

- Table 3.3.2.1-1, Note (i), "If the as-found channel setpoint is not the nominal trip setpoint (NTSP), the channel is inoperable. The NTSP is specified in the COLR. The methodology used to determine the NTSP is specified in the Bases," would be added for SR 3.3.2.1.7 for Functions 1.a, 1.b, and 1.c.

  Note (i) requires a determination if the as-found value is not the NTSP. If the as-found value is not the NTSP, the channel would be declared inoperable and the appropriate Action statement of LCO 3.3.2.1 would be entered. The note also requires that the NTSP is to be specified in the COLR and the methodology used to determine the NTSP to be specified in the Bases.

  In a response to an NRC staff RAI, the licensee explained that the RBM setpoints do not include as-found tolerances due to the digital nature of the device. If an as-found setpoint is not the NTSP, no as-found setpoint evaluation is needed because the channel would be declared inoperable. This requirement is included in proposed TS Note (i) and is acceptable to the NRC staff because it meets the requirements of 10 CFR 50.36 as discussed in RIS 2006-17.

  For an RBM channel to be declared operable after calibration, the as-left setpoint must be the same as the NTSP. The NRC staff finds this acceptable because the requirements of 10 CFR 50.36 are met by including the location of the NTSP and the setpoint methodology used in proposed Note (i).

- SR 3.3.2.1.4 would be revised. Related to the addition of the three power-dependent RBM functions (TS 3.3.2.1, Table 3.3.2.1-1, Functions 1.a, 1.b, and 1.c), SR 3.3.2.1.4 would be revised to require that these RBM power ranges are enabled at the appropriate power levels. Namely, the applicable limits (i.e., Low Power Range limit, Intermediate Power Range limit, and High Power Range limit) would be effective when the power is at or above the lower power limit for each range (the limit on permitted local power increase becomes more restrictive as the RBM power range increases). Function 1.a would not be bypassed when the APRM Simulated Thermal Power is ≥ 28% and < 63% RTP and a peripheral control rod is not selected; Function 1.b would not be bypassed when the APRM Simulated Thermal Power is ≥ 63% and < 83% RTP and a peripheral control rod is not selected; and Function 1.c would not be bypassed when the APRM Simulated Thermal Power is ≥ 83% RTP and a peripheral control rod is not selected. The NRC staff finds this acceptable because the bypass power levels would be consistent with the calculations in GE document 0000-0053-1006 NMP2 A-M-T506-RBM-Calc-2006, Revision 0 and with the recommendations of the NUMAC PRNM LTR.

3.12.5 TS 3.4.1, Recirculation Loops Operating

LCO 3.4.1d would be deleted. LCO 3.4.1d requires:

> "LCO 3.3.2.1, "Control Rod Block Instrumentation," Function 1.a (Rod Block Monitor – Upscale), Allowable Value of Table 3.3.2.1-1 is reset for single loop operation."

The change from the current flow-dependent RBM function and AV to three power-dependent functions and AVs in TS 3.3.2.1 would eliminate the need to maintain flow-dependent RBM – Upscale AVs for two loop and single loop operation. Therefore, the LCO 3.4.1d restriction to reset the RBM – Upscale AV when entering single loop operation is unnecessary, and the NRC staff agrees that it may be removed.

Editorial changes would be made to items b. and c. to reflect the deletion of item d.

3.12.6 TS 5.6.5, Core Operating Limits Report (COLR)

Item 5 would be revised from:

> "Control Rod Block Instrumentation Setpoint for the Rod Block Monitor – Upscale Function Allowable Value for Specification 3.3.2.1."

To:

> "The Allowable Values, NTSPs, and MCPR conditions for the Rod Block Monitor – Upscale Functions for Specification 3.3.2.1."

With the changes to TS 3.3.2.1, Table 3.3.2.1-1, Function 1.a, the creation of new Functions 1.b and 1.c, and the addition of Note (h), this change in the COLR reporting requirements becomes an administrative change and is, therefore, acceptable to the NRC staff.

3.12.7 Conclusion for ARTS/MELLLA - Related TS Changes

The review of TS changes was performed to evaluate the changes that would be required to support the ARTS/MELLLA implementation at NMP2. This review covered the ARTS/MELLLA application for the CLTP.

Based on its review, the NRC staff concludes that the proposed TS changes are acceptable because the safety analyses supporting actual operation in the ARTS/MELLLA regimes at the CLTP have been reviewed as acceptable, and the NRC staff concludes that operation will not endanger the public health and safety.

The NRC staff also concludes that the ARTS/ MELLLA logic changes are consistent with the NRC staff-approved guidance in the NUMAC PRNMS LTR and no exceptions have been taken to the safety bases for the NUMAC PRNMS LTR.

3.13 LSSS

In an RAI to the licensee, the NRC staff requested information to support its assessment of the acceptability of the AV changes described in the LAR, and of issues in NRC's letter to the Nuclear Energy Institute Setpoints Methods Task Force dated September 7, 2005 (ADAMS Accession Number ML052500004), and in RIS 2006-17. The licensee's November 2, 2007, response is discussed below.

3.13.1 SL-Related LSSS

The licensee identified that the TS 3.3.2.1, Table 3.3.2.1-1, new RBM power-dependent functions (1.a, Rod Block Monitor – Low Power Upscale, 1.b, Rod Block Monitor – Intermediate Power Upscale, and 1.c, Rod Block Monitor – High Power Upscale) are the only TS functions related to ARTS/MELLLA implementation that are SL-related LSSS.

The three power-dependent rod block functions would be credited in the accident analysis with protecting the MCPR SL specified in TS 2.1.1.2 for an RWE event. An RWE event is designated as an AOO. AOO events can be mitigated with a highly reliable non-safety system. The RBM is a highly reliable system and is designed to prohibit erroneous withdrawal of a control rod so that local fuel damage does not occur. Although the RBM is not classified as safety-related, it is part of the PRNMS, which contains safety-related components. As such, the RBM is designed to maintain physical integrity and separation under all conditions so that it is highly unlikely that a failure in the RBM could prevent the safety-related components of the PRNMS from performing its safety functions.

The RBM is designed, manufactured, and qualified to the same standards as the PRNMS. Procurement and factory acceptance was in accordance with the PRNMS specification for both safety-related and non-safety-related equipment.

Based on the RBM robust design, as described above, and on the operability requirements for the RBM in TS 3.3.2.1, the NRC staff finds that there is reasonable assurance that the RBM will perform its necessary function to mitigate the consequences of an RWE event. The licensee stated that the AV and setpoints for the RBM power-dependent functions are calculated on a cycle specific basis using GE setpoint methodology. As noted previously, the GE setpoint methodology is described in NEDC-31336 P-A and has been approved by the NRC staff. The new RBM AVs were calculated using this methodology and the results are included in GE document 0000-0053-1006 NMP2 A-M-T506-RBM-Calc-2006, Revision 0.

Because the RBM power-dependent functions (Table 3.3.2.1-1, Functions 1.a, 1.b, and 1.c) are SL-related LSSS, the licensee would add Note (i) to the references to SR 3.3.2.1.7 for Functions 1.a, 1.b, and 1.c to implement the setpoint-related TS to meet the intent of RIS 2006-17. The acceptability of Note (i) was discussed previously in this SE.

3.13.2 Non-SL-Related LSSS

The AV for TS 3.3.1.1, Table 3.3.1.1-1, Function 2.b, APRM Flow – Biased Simulated Thermal Power – Upscale, which is related to ARTS/MELLLA implementation, is not a SL-related LSSS. Controls are in place to ensure that the APRM Flow-Biased Simulated Thermal Power - Upscale function will perform in accordance with applicable design requirements. The as-left trip settings would be controlled under procedures based on the licensee's Surveillance Test Program. As-found settings found outside acceptable tolerances would be addressed through the NMP2 10 CFR Part 50, Appendix B, Criterion XVI, corrective action program. Operability determinations are integral to the corrective action program. When the condition described in a condition report under the program represents an operability concern, an operability determination would be completed. The return of a degraded or nonconforming component to a fully-qualified status would be added under the corrective action program. Instrument reference accuracy would be used for the as-found and as-left tolerances. An as-left setting would be procedurally required to be within the required as-left tolerance. If the as-found setting is outside the required as-found tolerance, the device would be reset to within the as-left tolerance.

The licensee explained that the setpoint methodology used for the APRM flow- biased simulated thermal power setpoint is the same methodology that is used for the RBM setpoints. As such, the setpoints and the uncertainty analysis for APRM flow-biased simulated thermal power are determined using the same criteria and rigor as setpoints used to protect the SL-related LSSS.

The NRC staff finds that sufficient measures are in place, through implementation of these controls, to ensure that the associated setpoints are capable of performing their safety functions.

3.13.3 Conclusion for LSSS

The NRC staff finds the identification and bases of the SL-related LSSS being removed, altered, or added by this LAR meet the requirements of 10 CFR 50.36(c)(1)(ii)(A).

For setpoints that are not SL-related, the NRC staff finds that acceptable measures, controls, and procedures are in place to ensure that the associated instrument channels are capable of performing their specified safety functions in accordance with applicable design requirements and associated analyses.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20, and changes SRs. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (72 FR 28721). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

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