SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 163 TO FACILITY OPERATING LICENSE NO. NPF-57

PSEG NUCLEAR LLC

HOPE CREEK GENERATING STATION

DOCKET NO. 50-354

1.0 INTRODUCTION

By letter dated June 7, 2004 (Reference 1), as supplemented by letters dated February 18, May 20, June 16, July 8, August 3, September 23, and November 16, 2005, and February 6, 2006 (References 2 through 9, respectively), PSEG Nuclear LLC (PSEG or the licensee) submitted a request to amend the Technical Specifications (TSs) for the Hope Creek Generating Station (HCGS). The proposed changes would implement the Average Power Range Monitor, Rod Block Monitor Technical Specifications (ARTS) improvement program concurrently with the implementation of the Maximum Extended Load Line Limit Analysis (MELLLA) operating power-flow map in order to allow additional startup and operating flexibility from an expanded operating domain. To support these proposed changes, the licensee's submittal provided a HCGS-specific ARTS/MELLLA safety analysis report (A/MSAR), NEDC-33066P (Reference 10), prepared by the Nuclear Steam Supply System vendor, General Electric Nuclear Energy (GENE). The supplements dated February 18, May 20, June 16, July 8, August 3, September 23, and November 16, 2005, and February 6, 2006, did not expand the scope of the application as originally noticed and did not change the Nuclear Regulatory Commission (NRC or the Commission) staff's original proposed no significant hazards consideration determination (69 FR 55471).

HCGS is a boiling water reactor (BWR), 4-series reactor, and the rated thermal power (RTP) is 3339 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 power is achieved, periodic adjustments to core flow and control rod positions must be made to compensate for the reactivity changes due to fuel and burnable poison burnup, and changes in Xenon concentration. Factors currently restricting plant flexibility at HCGS in efficiently achieving and maintaining rated power include:

1. the currently-licensed allowable power/flow operating map; and,

2. the current Average Power Range Monitor (APRM) flow-biased flux scram and flow-biased rod block setdown requirements.

PSEG has proposed TS changes to address the above restrictions. The proposed changes are similar to those approved at other BWRs.

The fuel-dependent portions of the safety analyses are based on the Cycle 13 core reference loading pattern (RLP). The Cycle 13 RLP is comprised of approximately 600 irradiated (from one to three previous cycles) non-GE14 fuel assemblies (SVEA-96+) in the core, and 164 fresh GE14 fuel assemblies. For the fuel-dependent portions of the safety analyses, PSEG performed plant- and fuel-specific analyses, using NRC-approved methodologies, to justify operation in the ARTS/MELLLA condition. The non-fuel-dependent evaluations are based on the HCGS plant configuration and were submitted for review in advance of the fuel dependent analyses.

The NRC staff notes that the licensee has submitted an amendment application that, if approved, would allow implementation of an extended power uprate (EPU) at HCGS. That application is being reviewed separately by the NRC staff and is not addressed in this Safety Evaluation (SE). This SE is specifically limited to implementation of ARTS/MELLLA at RTP.

2.0 REGULATORY EVALUATION

The NRC staff considered the following requirements and guidelines in its review of the licensing application:

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, General Design Criterion (GDC) 10 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 are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs). The licensee stated that it will validate or, if required, update the assumptions and conclusions relative to fuel-dependent evaluations to ensure the requirements of GDC 10 continue to be met.

Part 50 of 10 CFR, Appendix A, GDC 12 requires that the reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed. The licensee stated that it would validate or, if required, update the assumptions and conclusions relative to fuel-dependent evaluations to ensure the requirements of GDC 12 continue to be met.

Part 50 of 10 CFR, Appendix A, GDC 50 requires that the reactor containment structure 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 any loss-of-coolant accident (LOCA). The licensee performed evaluations to demonstrate that all containment parameters stay within their design limits.

Section 50.46 of 10 CFR sets forth acceptance criteria for the performance of the emergency core cooling system (ECCS) following postulated LOCAs. Part 50 of 10 CFR, Appendix K describes required and acceptable features of the evaluation models used to calculate ECCS performance. Although the proposed amendment to implement ARTS/MELLLA does not explicitly involve changes to the ECCS, the requirements of 10 CFR 50.46(b) are pertinent to the evaluation of acceptability of the proposed amendment. The requirements of 10 CFR 50.46(b) for maximum fuel element cladding temperature, maximum cladding oxidation, and maximum hydrogen generation during a design-basis LOCA must still be satisfied. The

licensee stated that it would validate or, if required, update the assumptions and conclusions relative to fuel-dependent evaluations to ensure the requirements of 10 CFR 50.46 continue to be met.

Section 50.49 of 10 CFR establishes requirements for environmental qualification of electric equipment important to safety for nuclear power plants. NRC Generic Letter (GL) 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," addresses additional considerations related to changes in post-accident containment response. The licensee performed evaluations to demonstrate that containment parameters stay within their design limits for steamline breaks, feedwater line breaks, and reactor water clean up system breaks.

Section 50.62 of 10 CFR, in part, specifies the equivalent flow rate, level of boron concentration and boron-10 isotope enrichment required for BWR standby liquid control systems (SLCS). The licensee stated that it would validate or, if required, update the assumptions and conclusions relative to fuel-dependent evaluations to ensure the requirements of 10 CFR 50.62 continue to be met.

Section 50.36 of 10 CFR, "Technical Specifications," provides the regulatory requirements for the content required in a licensee's TSs. Section 50.36 of 10 CFR requires that the TSs will include surveillance requirements (SRs) to assure that the limiting conditions for operation (LCOs) will be met. The TSs are required to meet the provisions of 10 CFR 50.36. Section 50.36(c)(1)(ii)(A) of 10 CFR requires that the TSs include limiting safety system settings. This paragraph specifies that where a limiting safety system setting is specified for a variable on which a safety limit (SL) has been placed, the setting must be chosen so that automatic protective action will correct the abnormal situation before an SL is exceeded. Accordingly, limits for instrument channels that initiate protective functions must be included in the TSs. Setpoints found to exceed TS limits are considered a malfunction of an automatic safety system. Such an occurrence could challenge the integrity of the reactor core, reactor coolant pressure boundary, containment, and associated safety systems.

Regulatory Guide (RG) 1.105, "Setpoints for Safety-Related Instrumentation," 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. In addition, the NRC provided clarification on the requirements of 10 CFR 50.36 in its August 23 and September 7, 2005, letters to the Nuclear Energy Institute (NEI). These letters clarified that some instrument channels perform functions important to safety and are cited in the TSs, but do not directly support the protection of an SL. For such channels, the setpoint and operability considerations should be established and controlled with rigor equivalent to that for SL-related TSs; however, there is no regulatory requirement that the setpoints and operability be explicitly controlled via the TSs.

3.0 TECHNICAL EVALUATION

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 SLs. The currently-approved operating domain for HCGS is the Extended Load Line Limit Analysis map with increased core flow (ICF) to 105% rated core flow (RCF) (Reference 11). The proposed TS change reflects operation of HCGS in a region which is above the current rated rod line.

The power/flow operating map includes the operating domain changes for ARTS/MELLLA consistent with NRC-approved operating domain improvements for other BWRs. This performance improvement program expands the operating domain along the approximate 119% rod line, to 100% RTP at 76.6% RCF. This domain is defined in Figure 1-1 of the GENE A/MSAR (Reference 10). 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 to be performed as part of the standard cycle-specific reload analysis.

The function of the Rod Block Monitor (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 minimum critical power ratio (SLMCPR) ratio or the 1% cladding plastic strain limit) in the event of a rod withdrawal error (RWE) event.

The functions of the APRM system include:

1. generation of a trip signal to scram the reactor during core-wide neutron flux transients before exceeding the safety analysis design basis,

2. blocking control rod withdrawal whenever operation exceeds set limits in the operating map, prior to approaching the scram level, and

3. 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 were provided to achieve these requirements.

The proposed implementation of the ARTS/MELLLA improvement program will increase the plant operating efficiency by updating the thermal limits requirements to be consistent with current GENE methodology and from changes in instrumentation setpoints.

The ARTS improvement program includes changes to the current APRM system, which requires the TS changes described in Section 3.8 of this SE. The current HCGS TSs require that the APRM setpoint provide adequate margin for the SLs through the application of an adjustment to the flow referenced trip setpoint when the core maximum fraction of limiting power density (CMFLPD) is greater than, or equal to, the fraction of rated thermal power (FRTP). This requirement limits the maximum local power at lower core power and flows to a fraction of that allowed at rated power and flow. If the CMFLPD exceeds the FRTP, the flow-referenced APRM trips must be lowered (set down) to limit the maximum power that the plant can achieve. The basis for this “APRM trip setdown” requirement originated from the now obsolete Hench-Levy Minimum Critical Heat Flux Ratio thermal limit criterion (Reference 12), and provides conservative restrictions with respect to current fuel thermal limits. A subsequent update to the thermal limits requirements, which decreases the dependence on the local thermal hydraulic conditions, including the core peaking factors, was developed by GENE. The

resulting General Electric Thermal Analysis Basis critical power ratio (CPR) correlation model (Reference 13), which relies on bundle boiling length and exit quality, was reviewed and approved by the NRC staff.

The objective of the APRM improvements is to justify removal of the APRM trip setdown. Since the elimination of the APRM gain and setpoint requirement can potentially affect the fuel thermal-mechanical integrity and the ECCS LOCA performance, the staff reviewed the acceptability of these changes. The following criteria are used to assure satisfaction of the applicable licensing requirements to demonstrate acceptability of the APRM trip setdown requirement:

1. The SLMCPR shall not be violated as a result of any AOOs,

2. All fuel thermal-mechanical design bases shall remain within the licensing limits described in the GENE generic fuel licensing report (GESTAR-II), and

3. The peak cladding temperature and the maximum cladding oxidation fraction following a LOCA shall remain within the limits defined in 10 CFR 50.46.

The ARTS-specific changes are summarized here:

1. The requirement for setdown of the APRM scrams and rod blocks is deleted,

2. New power-dependent minimum critical power ratio (MCPR) adjustment factors, MCPR(p), are added,

3. New flow-dependent MCPR adjustment factors, MCPR(f), are added,

4. New power-dependent linear heat generation rate (LHGR) adjustment factors, LHGRFAC(p), are added,

5. New flow-dependent LHGR adjustment factors, LHGRFAC(f), are added, and

6. The affected TS SRs, LCOs, and the associated Bases are modified or deleted, as required.

The NRC staff reviewed the safety analyses and systems response evaluations performed by GENE to justify HCGS operation in the expanded MELLLA region, as discussed in Reference 10. The plant-specific, fuel independent evaluations, such as containment response, were performed based on the current hardware design and applicable plant geometry for HCGS. The licensee performed fuel-dependent analyses including the limiting AOOs, the MCPR calculations, and the reactor vessel overpressure protection analysis. The current cycle (Cycle 13) core RLP was the basis for this evaluation. These analyses are to be performed each operating cycle as part of the standard reload design process, outlined in the current version of GESTAR-II (Reference 14).

3.1 Method of Analysis

The analyses that were used to justify operation with the ARTS improvement and the MELLLA power/flow operating map for a mixed-core of SVEA-96+ and GE14 fuels are based on GENE computer codes, methodologies, and applicable industry standards, which are discussed in the A/MSAR, associated references, and in response to the NRC staff's requests for additional information (RAIs). Table 1-1 of the HCGS A/MSAR (Reference 10) lists the GENE computer codes used in the safety analyses.

The SVEA-96+ fuel type was treated as a unique fuel design in the standard GENE design and licensing process. Sufficient base data about the SVEA-96+ were obtained by the licensee to model the fuel type's neutronic and thermal-hydraulic characteristics. The ISCOR code (Reference 15) is used to establish the thermal-hydraulic compatibility of the SVEA-96+ fuel loaded into a core with GE14 fuel. ISCOR is the code that implements the NRC-approved methodology to perform steady-state thermal-hydraulic analyses of a nuclear reactor core. Thermal performance calculations were carried out using the GEXL14 critical quality-boiling length correlation for GE14 fuel (Reference 16), and thermal performance calculations for SVEA-96+ fuel design were carried out using the GEXL80 critical quality-boiling length correlation (Reference 17), subject to the restrictions contained in the applicable NRC SEs. Given the detailed modeling of each fuel design using the ISCOR methodology, the key thermal-hydraulic performance parameters, such as pressure drop and channel flow, were validated for the mixed core to be comparable. Examples of the thermal-hydraulic compatibility for various combinations of the SVEA-96+ and GE14 fuel were provided by the licensee in response to the NRC staff's RAIs (Reference 8).

The neutronic characteristics for the HCGS application were determined using the TGBLA06/PANAC11 methodology (NRC approved per GESTAR II, Amendment 26). As mentioned earlier, both the fuel types were modeled in detail with each fuel type's respective thermal-hydraulic characteristics as well as their respective GEXL correlations. Thus, whether the core has multiple reloads of SVEA-96+ and smaller fractions of GE14, or the converse, each and every fuel type was specifically modeled and analyzed in its defined core location.

As discussed by the licensee in response to the NRC staff's RAI, the compatibility between fuel types and demonstration of the modeling capability for mixed-core situations was performed by a multi-cycle benchmark of HCGS SVEA-96+ transition cores. This benchmark comparison of key parameters such as hot and cold eigenvalue, power distribution limits and traversing in-core probe system (TIPS) measurements demonstrated that the SVEA-96+ fuel design was compatible with the integrated TGBLA06/PANAC11 methodology (Reference 10). In addition to multi-cycle benchmark comparisons, lattice-specific benchmarking for the SVEA-96+ design was conducted by the licensee. At one or more specific combinations of lattice design, exposure, void content, historical void content, boron inventory, the lattice reactivity, and local pin fission density were compared between TGBLA and the MCNP Monte Carlo neutron transport program. The result of these investigations was that SVEA-96+ was modeled acceptably by TGBLA. Combining this lattice-specific benchmark scope with the integrated TGBLA/PANACEA, HCGS's specific multi-cycle benchmark demonstrated a compatible process relative to the ability of the GENE methods to model the SVEA-96+ fuel in addition to the existing capability to model the GE14 fuel.

The licensee continues to perform cycle-exposure accounting, which examines hot and cold

eigenvalue trending and comparison of simulated thermal limits to monitored thermal limits at HCGS. This will provide on-going confirmation that the mixed-core of GE14 and SVEA-96+ continues to be compatible from a neutronic perspective.

The ARTS transient analyses were performed at the RTP plant conditions for the current Cycle 13 core configuration, using the GENE standard reload licensing methodology described in the GESTAR-II documentation (Reference 14). The HCGS plant-specific evaluations were performed to establish plant-unique MCPR(f), LHGR, and maximum average planar linear heat generation rate (MAPLHGR) limits. Added conservatisms were included, which are expected to allow future reloads of GE14 and SVEA-96+ fuel design.

The NRC staff finds the licensee's method of analysis for the HCGS MELLLA operation acceptable. The staff is currently reviewing GENE methods for other applications such as a new expanded operating domain and for EPUs. Should the staff identify any issues with the GENE methods that would bear on its conclusions in this SE, the staff will address the issue on a plant-specific basis.

3.2 Fuel Thermal Limits

Extensive transient analyses at a variety of power and flow conditions were performed during original development of the ARTS improvement program. These evaluations are applicable for operation in the ARTS/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 included evaluations representative of a variety of plant configurations and parameters such that the conclusions drawn from the studies would be applicable to all BWRs. The database was utilized to develop a method of specifying plant operating limits (MCPR, and LHGR or MAPLHGR), which ensures that margins to fuel SLs are equal to, or larger than, those applied currently.

The NRC staff reviewed the effects of operation along the higher MELLLA rod line at the RTP on the thermal limits and the thermal limits management with the ARTS power- and flow-dependent limits, which are covered in the A/MSAR. The potentially limiting AOOs and accident analyses were evaluated to support HCGS operation with the ARTS off-rated limits, as well as operation in the MELLLA region for the current operating Cycle 13.

The core-wide AOOs included in the current HCGS Cycle 13 reload licensing analyses and the HCGS Updated Final Safety Analysis Report (UFSAR) were re-examined by the licensee for operation in the ARTS/MELLLA region (including off-rated power and flow conditions). The following events were considered potentially limiting in the ARTS/MELLLA region and were reviewed as part of the ARTS program development:

1. generator load rejection with no bypass (LRNBP) event;

2. turbine trip with no bypass (TTNBP) event;

3. feedwater controller failure (FWCF) maximum demand event;

4. loss of feedwater heating (LFWH) event;

5. fuel loading error (FLE) event;

6. idle recirculation loop start-up (IRLS) event; and

7. fast recirculation flow increase (FRFI) event.

The initial ARTS/MELLLA assessment of these events for all BWR 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. The LFWH evaluation at 76.6% RCF for HCGS Cycle 13 showed that there is a large margin in operating limit minimum critical power ratio (OLMCPR) for the LFWH event compared to the LRNBP event (1.20 for the LFWH versus 1.43 for the LRNBP). Considering that the LFWH event tends to become less limiting as the power decreases (less feedwater to be affected by loss of heating), the LFWH event was not considered in the determination of the off-rated limits. The FLE event is most limiting at maximum power; thus, this event was also not considered in the determination of the off-rated limits. The other two events (IRLS and FRFI) are generally most limiting at off-rated conditions. However, even when originated from their most limiting off-rated condition during ARTS/MELLLA operation, the IRLS and FRFI events are less limiting than those events that determine the generic off-rated multipliers and operating limits.

The generic evaluations determined that the power-dependent severity trends must be examined in two power ranges. The first power range is between rated power and the power level, P_{bypass}, where reactor scram on turbine stop valve closure or turbine control valve fast closure is bypassed. P_{bypass} for HCGS 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 (applicability of TS 3/4.2, "Power Distribution Limits").

Generic power-dependent MCPR, and LHGR or MAPLHGR limits (in terms of multipliers on the plant's rated operating limits) were developed for use in the power range between rated power and P_{bypass}. Below P_{bypass}, the OLMCPR is specified while the MAPLHGR (or LHGR) retains the use of a multiplier on the rated limits. Generic flow-dependent MCPR and MAPLHGR (or LHGR) limits were also developed from the ARTS database.

HCGS-specific analyses were performed to confirm the applicability of the generic power-dependent limit multipliers [K(p), LHGR(p), and MAPLHGR(p)] above P_{bypass}. HCGS-specific evaluations were also performed between P_{bypass} (30%) and 25% power to establish HCGS-unique MCPR, LHGR, and MAPLHGR limits. These limits have been confirmed for initial application of the GE14 fuel type at HCGS as well as the SVEA-96+ fuel in the Cycle 13 core, including margin to bound future fuel cycles. HCGS specific evaluations were also performed to establish the flow-dependent MCPR, LHGR, and MAPLHGR limits.

The AOO analyses were performed for the current operating Cycle 13 RTP conditions with the MELLLA operating power/flow statepoints, generating operating limits for the current Cycle 13 core GE14 and SVEA-96+ fuel configuration. For AOOs, cycle-specific analyses are performed for the limiting transients. These transient analyses use the cycle-specific nuclear and thermal-hydraulic characteristics of the reload core to establish the rated and off-rated power operating

limits for the fuel types that comprise the reload core (whether a mixed core of SVEA-96+ and GE14, or a full core of GE14). These transient analyses also consider the ARTS/MELLLA operating domain for establishing initial conditions for the transient initiation. Either fuel type may be limiting for a specific analysis condition.

The partial ARTS improvement implementation at HCGS will not require the original ARTS hardware change to the RBM that provided protection for an off-rated RWE event. Therefore, evaluation of the HCGS RWE event is performed without taking credit for the mitigating effect of the flow-biased RBM setpoints, and the resulting off-rated OLMCPR values are for the unblocked RWE event. This is consistent with the implementation of the partial ARTS program at the other BWRs, such as Vermont Yankee Nuclear Power Station. The cycle-specific (Cycle 13) analysis was performed for HCGS's plant-specific, power-dependent RWE OLMCPR value. If, for future reload cycle operating conditions, the unblocked RWE event OLMCPR values are too restrictive, the RBM setpoint adequacy would be readdressed by the licensee.

Since 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.

3.3 Vessel Overpressure

The main steam isolation valve (MSIV) closure with a flux scram (MSIVF) event is used to determine compliance with the industry standard American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). A reload cycle-specific calculation was performed at 102% of RTP with core flows from 76.6% to the maximum licensed core flow (105% of rated). The implementation of ARTS/MELLLA does not affect the maximum core flow case since operation at the ICF is currently licensed. The new minimum flow allowed by MELLLA (76.6%) is slightly bounding and provides acceptable results.

3.4 Thermal-Hydraulic Stability

Stability criteria are established in GESTAR-II to demonstrate compliance with the GDC 12 requirements in order to assure that specified acceptable fuel design limits (i.e., SLMCPR) are not exceeded. The analysis and methods used to demonstrate compliance with the stability acceptance criteria are documented in NEDO-31960A (Reference 18).

PSEG has licensed Option III (Reference 18) as the long-term solution, and has an approved TS for the Option III hardware. The Option III hardware has been installed and connected to the reactor protection system. In the event that the oscillation power range monitor (OPRM) system is declared inoperable, HCGS will operate under the Boiling Water Reactor Owners' Group (BWROG) Guidelines for Stability Interim Corrective Action, as described in Reference 19.

The Option III solution combines closely-spaced local power range monitor (LPRM) detectors into "cells" to effectively detect either core-wide or regional (local) modes of reactor instability. These cells are termed OPRM cells and are configured to provide local area coverage with multiple channels. Plants implementing Option III install new hardware to combine the LPRM signals and to evaluate the cell signals with instability detection algorithms. The Period-Based

Detection Algorithm is the only algorithm credited in the Option III licensing basis. Two defense-in-depth algorithms, referred to as the Amplitude Based Algorithm and the Growth Rate Based Algorithm, offer a high degree of assurance that fuel failure will not occur as a consequence of stability-related oscillations.

Stability Option III provides SLMCPR protection by generating a reactor scram if a reactor instability that exceeds the specified trip setpoint is detected. The acceptable setpoint is determined for each operating cycle per the NRC-approved methodology (Reference 20). The Option III stability reload licensing basis calculates the limiting OLMCPR required to protect the SLMCPR for both steady-state and transient stability events as specified in the Option III methodology. These OLMCPRs are calculated for a range of OPRM setpoints for MELLLA operation. Selection of an appropriate instrument setpoint is then made based upon the OLMCPR required to provide adequate SLMCPR protection. This determination relies on the DIVOM curve (Delta CPR Over Initial CPR Versus Oscillation Magnitude) to determine an OPRM setpoint that protects the SLMCPR during an anticipated instability event for the MELLLA operation.

The licensee stated that sufficient data about the SVEA-96+ fuel design was obtained to model the fuel type's neutronics and thermal-hydraulic characteristics. In addition, as stated earlier, an NRC-approved critical power correlation for the SVEA-96+ fuel (GEXL80) was developed to provide the correct CPR response (Reference 16). This allowed GENE to use the TRACG code, which is the GENE proprietary version, to establish DIVOM curve. A plant- and cycle-specific DIVOM slope is developed in accordance with the BWROG Regional DIVOM Guideline (Reference 21).

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

3.5 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. The MAPLHGR operating limit is based on the most limiting LOCA and ensures compliance with the ECCS acceptance criteria in 10 CFR 50.46.

The HCGS SAFER/GESTR-LOCA analysis for low core flow conditions in the MELLLA region was evaluated and documented in Reference 22. The Reference 22 analysis is applicable to the HCGS current GE14 and SVEA-96+ fuel designs operating at RTP conditions with no change in plant configuration. **[[**

]] The Appendix K PCTs **[[** **]]** are 1351 EF for GE14 **[[** **]]** and 1530 EF for SVEA-96+ **[[** **]]** The Licensing Basis PCTs are below 1600 EF for

both fuel types; therefore, there is significant margin to the 2200 °F PCT limit in the MELLLA region.

The PCT for a large recirculation line break is affected by MELLLA because the core flow is reduced in the MELLLA range, which leads to earlier boiling transition at lower elevation in the fuel bundle. For small breaks, however, the fuel remains in nucleate boiling until uncovery; therefore, MELLLA does not have an adverse effect on the small-break LOCA response. MELLLA has a negligible effect on compliance with the other acceptance criteria of 10 CFR 50.46. The NRC staff has, therefore, determined that no additional operating restrictions would be required for ARTS/MELLLA operation at the RTP, since the determination of the sensitivity of the ECCS-LOCA evaluations to operation in the MELLLA domain shows compliance with the acceptance criteria.

3.6 Anticipated Transient without Scram (ATWS)

The basis for the current ATWS requirements is 10 CFR 50.62. This regulation includes requirements for an ATWS recirculating pump trip, an alternate rod insertion (ARI) system, and an adequate SLCS injection rate.

The NRC staff reviewed the HCGS-specific analysis that was performed using the approved licensing methodology (Reference 23) to demonstrate compliance with 10 CFR 50.62 ATWS requirements. The analysis assumed the RTP with the minimum MELLLA core flow (76.6% of RCF), which is the limiting operating condition. The limiting ATWS events, MSIV closure and pressure regulator failure open (PRF), were re-evaluated with ARI assumed to fail, requiring SLCS injection. The adequacy of the margin to the SLCS relief valve lifting, as described in NRC Information Notice 2001-13, “Inadequate Standby Liquid Control System Relief Valve Margin,” was included in this assessment. One safety/relief valve (SRV) was assumed to be out of service, which was specified as the SRV with the lowest set point.

The limiting ATWS event for HCGS is the PRF. The maximum SLCS pump discharge pressure and timing depend primarily on the SRV set points. The maximum SLCS pump discharge pressure during the limiting ATWS event is approximately 1258 psig. This value is based on a peak reactor vessel upper-plenum pressure of 1164 psig that occurs during the time the SLCS was analyzed to be in operation during the limiting PRF event at the beginning of cycle.

For a nominal SLCS relief valve set point of 1400 psig, there is a margin of approximately 142 psi between the peak SLCS pump discharge pressure and the relief valve nominal set point. Therefore, there is adequate margin to prevent the SLCS relief valve from lifting (per NRC Information Notice 2001-13).

The NRC staff concludes, based on its review of the above analyses, that HCGS meets the ATWS mitigating features stipulated in 10 CFR 50.62 and that the results of the ATWS analyses for MELLLA operation at the RTP would meet the ATWS acceptance criteria for the current operating Cycle 13.

3.7 Effects of Increased Operating Domain

Operation in the MELLLA domain can result in a decrease in the temperature of the fluid in the downcomer. This can cause changes in the containment pressure and temperature response following a LOCA. NEDC-33006P states that the effect of the expanded operating domain on containment response must be evaluated on a plant-specific basis. As part of its ARTS/MELLLA application, the licensee evaluated the containment response for the short-term containment pressure and temperature, containment dynamic loads (subcompartment pressurization), and equipment qualification (including GL 96-06). The HCGS containment-related analyses were documented in Section 8, “Containment Response,” of NEDC-33066P (Reference 10).

The HCGS principle design criteria (PDC) are listed in Section 1.2.2 of the HCGS UFSAR. UFSAR Section 6.2.1.1, “Pressure Suppression Containment,” discusses the containment design basis and includes three functional capabilities related to this review:

8. The containment has the capability to maintain its functional integrity during and following peak transient pressures and temperatures that occur following any postulated LOCA. The LOCA includes the worst single failure (which leads to maximum containment pressure and temperature) and is further postulated to occur simultaneously with a loss of offsite power and a safe shutdown earthquake.

9. The containment, in combination with other accident mitigation systems, limits fission product leakage following a postulated DBA such that offsite doses remain below the criteria in 10 CFR 50.67, “Accident Source Term.”

10. The containment can withstand coincident fluid jet forces associated with the flow from the postulated rupture of any pipe within the containment.

To demonstrate that the above criteria are safisfied for MELLLA’s extended operating domain, the licensee evaluated the containment short-term pressure and temperature response, including the pressure, temperature, and humidity levels in subcompartments.

Short-term containment pressure and temperature response remaining below the design values stated in UFSAR Table 6.2-1, “Containment Design Parameters,” ensures that design basis functional capabilities one and two (above) are met, and demonstrates that the HCGS PDC are maintained. Containment dynamic loads remaining below the design values stated in UFSAR Appendix 6B, “Subcompartment Differential Pressure Considerations,” ensures that design basis functional capabilities one, two and three are met, and demonstrates that the HCGS PDC are maintained. Temperature, pressure, and humidity levels in containment compartments remaining below the values in the HCGS environmental design criteria (EDC) provides adequate assurance that safety-related systems can perform their safety functions under accident conditions.

3.7.1 Short-Term Containment Response

When evaluating the short-term containment pressure and temperature response for the design-basis LOCA, the licensee used the same assumptions as those used in the current UFSAR analysis. The licensee's sensitivity study showed that, for a constant feedwater inlet temperature, the limiting condition was 102% RTP with 100% RCF. This reactor operating condition is analyzed as part of the UFSAR and the plant unique load definition (PULD) analyses. Therefore, the current UFSAR and PULD analyses remain limiting with regard to the containment response.

The licensee's sensitivity study also showed that, for a reduced feedwater temperature, the limiting case was 102% RTP and 76.6 % RCF. This case results in an increase in the peak drywell pressure of 0.2 psi and an increase in the drywell-to-wetwell peak differential pressure of 0.1 psi as compared to the constant feedwater temperature case. The peak drywell pressure and the drywell-to-wetwell peak differential pressure for this case are 0.1 psi above the values in the current UFSAR. The licensee concluded that the effects of MELLLA on the reduced feedwater temperature case are negligible. The NRC staff reviewed the licensee's basis for this conclusion and determined that 1) the methods used by the licensee are conservative (Reference 25), and 2) the changes in the peak values are within the conservatism of the analytical methods. Therefore, the staff agrees with the licensee's conclusions that the 0.1 psi calculated changes are negligible.

3.7.2 Containment Dynamic Loads

The licensee evaluated the containment dynamic loads using the short-term containment pressure and temperature response. The licensee's evaluation showed that the effects of MELLLA on the containment are either bounded by current UFSAR analyses and remain within the original design basis, or the changes are negligible. Therefore, the licensee concluded that there is no significant effect on the containment LOCA dynamic loads. The NRC staff notes that there is considerable conservatism in the methods of analyses, as noted above. The staff agrees with the licensee's conclusion and finds there is reasonable assurance that the expanded operating domain will not adversely affect the HCGS containment dynamic loads response to the design-basis LOCA.

3.7.3 Environmental Design

The licensee compared each compartment peak pressure, temperature, and humidity level to the EDC and found two cases where the EDC are exceeded. The peak temperature exceeded the EDC value of 217 EF by 0.4 EF in one case and 0.3 EF in another case. The licensee concluded these differences are negligible with respect to the qualification of equipment in the affected compartments. The NRC staff agrees with the licensee's conclusions and finds there is reasonable assurance that the expanded operating range will not adversely affect the equipment qualification. Since the changes in the temperatures and pressures are not significant, the staff also finds that HCGS continues to meet the intent of GL 96-06.

3.7.4 Reactor Asymmetric Loads

One of the contributors to the reactor asymmetric loads evaluation is the reactor vessel annulus pressurization resulting from a pipe break in this region. For HCGS, the current method used to generate the mass and energy releases from the pipe break into the annulus at HCGS is described in NEDO-24548 (Reference 26), and the current method used to determine the pressure response in the annulus region is the COPDA code (Reference 27).

The licensee stated that the mass and energy profile calculated for the MELLLA expanded operating domain using the method described in NEDO-24548 exceeded the HCGS design calculation. The method develops the mass and energy releases from a composite of an assumed finite opening time pipe break and an instantaneously opening pipe break, and adds an additional layer of conservatism by quickly ramping the mass released to a value that exceeds the maximum expected flowrate and quickly ramping the energy associated with the fluid to the maximum expected value.

The licensee proposed to use the approved LAMB code (Reference 28) to calculate the mass and energy release from the pipe break in order to remove the additional layer of conservatism in the NEDO-24548 method. LAMB accounts for the pipe break separation time to calculate the mass end energy release rates, but ignores the fluid inertia effect such that the results are conservative. The NRC staff has previously accepted LAMB for licensing applications for the power/flow map extension associated with EPUs (Reference 29) This is the first time that LAMB has been proposed for use in the annulus pressurization evaluation, but this application is similar to the approved use of LAMB for the short-term containment pressure response described in NEDC-32424-P-A (Reference 29). For the short time period of interest, the mass and energy release calculated with LAMB are not impacted by the downstream pressure because the flow is choked (e.g., differential pressure is large enough that the flow cannot be further increased by lowering the downstream pressure). The staff finds it acceptable for the licensee to use LAMB for the mass and energy release inputs to the annulus pressurization evaluation, because LAMB is also conservative for this evaluation. The additional conservatism in the NEDO-24548 enveloping calculation, when compared to the LAMB, is about 20% more mass and 15% more energy released over the one-second period of interest.

The licensee also proposed to use the COMPARE-MOD 1 code (Reference 30) as a replacement for COPDA for the HCGS annulus pressurization evaluation. COMPARE-MOD 1 and COPDA are similar, and were developed for applications like the annulus pressurization evaluation.

COPDA is used to predict pressure and temperature histories in containment subcompartments following a postulated pipe failure. The basic assumption of the code is that the steam-water-air mixture behaves homogeneously. This assumption is utilized in the temperature-pressure calculations, as well as the flow rate calculations. Junction flows are based on the Moody equation.

COMPARE-MOD1 performs transient analysis of the thermodynamic conditions in zero velocity (or stagnant) volumes connected by flowing junctions. Volume thermodynamics and junction flows are for homogeneous mixtures of steam, two-phase water, three perfect gases, or

combinations of the above. Vent flow can be based on the Moody equation, compressible polytropic orifice flow, and an incompressible sub-element inertial relationship. Variable area doors and heat sinks can be modeled. COMPARE-MOD1 was written to perform transient subcompartment analysis.

The licensee chose options in COMPARE-MOD 1 to match the current licensing basis and simulate COPDA. The licensee used the option for thermodynamic equilibrium of the air-steam mixture in each node, the option for inertial flow in the flow paths (junctions) connecting nodes, and the option for no heat-transfer to structures. The COMPARE-MOD 1 nodalization used the same annulus region nodal volumes, initial conditions, junction flow path areas, flow loss coefficients, and inertia terms as those used in the original COPDA evaluation. The COMPARE-MOD 1 analysis was governed by the licensee's Nuclear Quality Manual, which prescribes a Nuclear Quality Management System based on the regulatory criteria and requirements of 10 CFR 50 Appendix B, American National Standards Institute (ANSI) ANSI-N45 Series, and ANSI/ASME Code NQA-1 Standards.

The licensee provided a comparison study of the differential pressures between various nodes as a benchmark of the COMPARE-MOD 1 calculation to the UFSAR calculation using COPDA. The differential pressure is used to evaluate the dynamic loads on the structures. The peak pressure differences between nodes were in close agreement with the COPDA results. The COPDA results were about one to two psi greater near the pipe break location during the period of interest (i.e., when the dynamic loads are the largest). The net force comparison showed the COPDA results to be about 7% greater than the COMPARE-MOD 1 results. The staff finds the licencee's proposal to use COMPARE-MOD 1 code for the HCGS annulus pressurization study acceptable because the modeling features used are similar to those used in the COPDA code, and the sensitivity studies performed by the licensee demonstrate close agreement between the two codes.

3.8 TS Changes for ARTS/MELLLA

Section 50.36 of 10 CFR provides the regulatory requirements for the content required in TSs. Section 50.36 of 10 CFR requires that the TSs will include SRs to assure that the LCO will be met. The NRC staff reviewed the proposed changes to the HCGS TSs that are identified in the licensee's submittal. The changes include deletion of the current setdown requirements, and new power- and flow-dependent MCPR and MAPLHGR limits. The proposed TS changes include the following:

TS Table 2.2.1-1: The APRM flow-biased simulated thermal power - upscale trip set point was revised.

TS Table 3.3.6-2: The APRM flow-biased neutron flux - upscale rod block trip set point was revised.

TS Table 3.3.6-2: The RBM upscale flow biased trip set points were revised.

TS 3/4.2.2: "APRM Set points," which includes requirements for flow-biased APRM scram and rod block trip set point setdown, and the associated TS Bases, were deleted.

The following additional changes were made to reflect the deletion of TS 3/4.2.2:

1. References to TS 3/4.2.2 were deleted from TS 3/4.4.1 Actions a.2 and a.3, and from footnotes to TS Tables 4.3.1.1-1 and 3.3.6-2.

2. Reference to the Maximum Fraction of Limiting Power Density was deleted from the specified conditions in the Applicability for LCO 3.3.7.7.

3. Definitions for “Core Maximum Fraction of Limiting Power Density,” “Fraction of Limiting Power Density,” “Fraction of Rated Thermal Power,” and “Maximum Fraction of Limiting Power Density” were deleted from TS Section 1.0.

4. References to APRM trip set point adjustments were deleted from TS Bases 2.2.1.

5. TS 3/4.2.1, “Average Planar Linear Heat Generation Rate,” TS 3/4.2.3, “Minimum Critical Power Ratio,” TS 3/4.2.4, “Linear Heat Generation Rate,” and associated TS Bases were revised, as appropriate, to include a description of power- and flow-dependent thermal limits.

6. The TS Index was revised.

These changes allow the implementation of the thermal limits portion of the GENE ARTS improvement protection program and the MELLLA expanded operating domain. The safety analyses presented examined the same areas as previous ARTS and MELLLA submittals, which have been reviewed by the NRC staff. The methods used have been previously approved and the results of the analyses fall within accepted limits. The staff concludes that the results submitted by the licensee justify the proposed TS changes to HCGS for operation at the RTP, based on the analyses reviewed and compared with the prior approvals.

The proposed TS changes are also consistent with the requirements of NUREG-1433, “Standard Technical Specifications - General Electric Plants, BWR/4,” Revision 2. The NRC has previously approved similar amendments for plants, such as LaSalle County Station, Units 1 and 2 (References 31 and 32), Dresden Nuclear Power Station, Units 2 and 3 (Reference 33), Quad Cities Nuclear Power Station, Units 1 and 2 (Reference 34), and Vermont Yankee Nuclear Power Station (Reference 35).

The NRC staff evaluated the set point changes in TS Table 2.2.1-1 and TS Table 3.3.6-2, as described above. In its application, the licensee stated that the APRM and RBM functions being modified are not credited in any safety licensing analyses or for accident mitigation. The licensee indicated that these functions are part of the original BWR plant design and established TSs. They are not credited in the safety analyses and are being retained in the TSs for operational reasons rather than because of safety-related necessity. Regarding the set point methodology issues discussed in the August 23, and September 7, 2005 letters to the NEI, the NRC staff reviewed the licensee’s application and concluded that the revised set points and allowable values are not related to variables upon which an SL has been placed. The licensee also indicated that, at the conclusion of each surveillance test, the channel set points are restored to values that are within the specified acceptance criteria of the nominal set point

values. In addition, the licensee indicated that if the as-found value of a set point is found to be nonconservative relative to the associated allowable value, the channel will be declared inoperable and the associated TS action statement will be invoked. Based upon the above, the staff concludes that there is reasonable assurance that the plant will operate in accordance with the safety analyses and that the operability of the instrumentation is ensured. Therefore, the staff finds that the proposed TS changes are acceptable.

The licensee also provided the associated TS Bases that reflect the proposed TS changes. The TS Bases changes are consistent with the licensee's proposed plant-specific TS changes, and the staff has no objections to the Bases changes presented in the licensee's application.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Jersey State Official was notified of the proposed issuance of the amendment. As indicated by letter dated April 22, 2005, 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. 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 (69 FR 55471). 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

1. Letter from Brothers, M. H., PSEG, to NRC, "Request for License Amendment ARTS/MELLLA Implementation Hope Creek Generating Station," dated June 7, 2004.

2. Letter from Barnes, G. P., PSEG, to NRC, "Supplement to Request for License Amendment ARTS/MELLLA Implementation Hope Creek Generating Station," dated February 18, 2005.

3. Letter from Barnes, G. P., PSEG, to NRC, “Response to Request for Additional Information Request for License Amendment ARTS/MELLLA Implementation Hope Creek Generating Station,” dated May 20, 2005.

4. Letter from Barnes, G. P., PSEG, to NRC, “Response to Request for Additional Information Request for License Amendment ARTS/MELLLA Implementation Hope Creek Generating Station,” dated June 16, 2005.

5. Letter from Barnes, G. P., PSEG, to NRC, “Supplement to Request for License Amendment Errata and Addenda for NEDC-33066P Revision 2 ARTS/MELLLA Implementation Hope Creek Generating Station,” dated July 8, 2005.

6. Letter from Barnes, G. P., PSEG, to NRC, “Response to Request for Additional Information Request for License Amendment ARTS/MELLLA Implementation Hope Creek Generating Station,” dated August 3, 2005.

7. Letter from Barnes, G. P., PSEG, to NRC, “Response to Request for Additional Information ARTS/MELLLA Implementation Hope Creek Generating Station,” dated September 23, 2005.

8. Letter from Barnes, G. P., PSEG, to NRC, “Response to Request for Additional Information ARTS/MELLLA Implementation Hope Creek Generating Station,” dated November 16, 2005.

9. Letter from Barnes, G. P., PSEG, to NRC, “Revised Technical Specification Bases ARTS/MELLLA Implementation Hope Creek Generating Station,” dated February 6, 2006.

10. “Hope Creek Generating Station, APRM/RBM/Technical Specifications / Maximum Extended Load Line Limit Analysis (ARTS/MELLLA),” NEDC-33066P, Rev. 2, GE Nuclear Energy, February 2005.

11. NEC-31487, “Increased Core Flow and Extended Load Line limit Analysis for Hope Creek Generating Station Unit 1 Cycle 2,” November 1987.

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13. “General Electric BWR Thermal Analysis Basis (GETAB) Data, Correlation, and Design Application,” NEDE-10958-P-A, November 1977.

14. “General Electric Standard Application for Reactor Fuel,” GE Nuclear Energy, NEDE-24011-P-A-14 and NEDE-24011-P-A-14-US, Revision 14, June 2000.

15. Global Nuclear Fuel, Licensing Topical Report, “General Electric Standard Application for Reactor Fuel,” NEDE-24011-P-A-15, Class III, September 2005.

16. GE Nuclear Energy, "GEXL14 Correlation for GE14 Fuel," NEDC-32851P, Rev.2, Class III, September 2001.

17. Global Nuclear Fuel, "GEXL80 Correlation for SVEA96+ Fuel," NEDC-33107P-A, Rev. 1, Class III, October 2004.

18. "BWR Owners Group Long Term Stability Solutions Licensing Methodology," NEDO-31960A and NEDO-31960A, Supplement 1, November 1995.

19. BWROG-94078, "BWR Owners' Group Guidelines for Stability Interim Corrective Action," June 6, 1994.

20. NEDO-32465-A, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications," August 1996.

21. Plant-Specific Regional Mode DIVOM Procedure Guideline," GE-NE-0000-0028-9714-R0, June 2004.

22. NEDC-33153P, "SAFER/GESTR-LOCA Loss of Coolant Accident Analysis for Hope Creek Generating Station," Revision 1, September 2004.

23. NEDC-24154P-A, "Qualification of the One Dimensional Core Transient Model (ODYN) for Boiling Water Reactors (Supplement 1 – Volume 4)," February 2000.

24. NEDC-33006P, Licensing Topical Report, "General Electric Boiling Water Reactor Maximum Extended Load Line Limit Analysis Plus," Revision 1, August 2003.

25. Letter, D. Eisenhut (NRC) to L. J. Sobon (GE), "Review of General Electric Topical Report NEDO-21052, 'Maximum Discharge Rate of Liquid-Vapor Mixtures from Vessels'," MFN-004-79, December 27, 1978 (with enclosed Topical Report Evaluation - NEDO-21052).

26. NEDO-24548, "Technical Description Annulus Pressurization Load Adequacy Evaluation," D. K. Sharma, General Electric Company, January 1979.

27. COPDA, "Subcompartment Pressure Analysis," BN-TOP-4, Revision 1, November 1977, Bechtel Power Corporation, San Francisco, CA.

28. NEDE-20566-P-A, "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50 Appendix K," September 1986.

29. NEDC-32424-P-A, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprates," GE Nuclear Energy, February 1999.

30. COMPARE-MOD 1: "A Code for the Transient Analysis of Volumes with Heat Sinks, Flowing Vents, and Doors" LA-7199-MS, March 1978.

31. LaSalle County Station, Units 1 and 2 - Issuance of Amendments (TAC Nos. M89631 and M89632), April 13,1995.

32. LaSalle County Station, Units 1 and 2 - Issuance of Amendments Regarding Power Uprate (TAC Nos. MA6070 AND MA6071), May 9, 2000.

33. Dresden Nuclear Power Station, Units 2 and 3 - Issuance of Amendments for Extended Power Uprate (TAC Nos. MB0844 and MB0845), December 21, 2001.

34. Quad Cities Nuclear Power Station, Units 1 and 2 - Issuance of Amendments for Extended Power Uprate (TAC Nos. MB0842 and MB0843), December 21, 2001.

35. Vermont Yankee Nuclear Power Station - Issuance of Amendment Re: Implementation of ARTS/MELLLA (TAC NO. MB8070), April 14, 2004.

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