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U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
REQUEST FOR CHANGE TO TECHNICAL SPECIFICATIONS
ARTS/MELLLA IMPLEMENTATION
HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354

References:
1. NRC letter, "Hope Creek Generating Station - Request for Additional Information Regarding Request for Change to Technical Specifications, Implementation of ARTS/MELLLA Operating Domain (TAC No. MC3390)," dated August 18, 2005
2. LR-N04-0062, "Request for License Amendment: ARTS/MELLLA Implementation," dated June 7, 2004
3. LR-N05-0032, "Supplement to Request for License Amendment: ARTS/MELLLA Implementation," dated February 18, 2005
4. LR-N05-0448, "Response to Request for Additional Information," dated September 23, 2005

This letter responds to the NRC's request for additional information (RAI) (Reference 1) regarding the license amendment request by PSEG Nuclear LLC (PSEG) in References 2 and 3 to revise the Technical Specifications (TS) for the Hope Creek Generating Station 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 amendment request also includes changes in the methods used to evaluate annulus pressurization (AP) and jet loads resulting from the postulated recirculation suction line break (RSLB).

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This letter forwards proprietary information in accordance with 10CFR 2.390. The balance of this letter may be considered non-proprietary upon removal of Attachment 1.

Reference 4 provided the responses to three of the NRC staff's questions. Attachment 1 to this letter provides responses to the remaining questions. Attachment 1 contains proprietary information as defined by 10 CFR 2.390. General Electric Company (GE), as the owner of the proprietary information, has executed the affidavit included in Attachment 1, which identifies that the attached proprietary information has been handled and classified as proprietary, is customarily held in confidence, and has been withheld from public disclosure. The proprietary information was provided to PSEG in a GE transmittal that is referenced by the affidavit. The proprietary information has been faithfully reproduced in the attached RAI responses such that the affidavit remains applicable. GE requests that the proprietary information be withheld from public disclosure in accordance with the provisions of 10 CFR 2.390 and 9.17. A non-proprietary version of the RAI responses also is provided in Attachment 2.

PSEG has determined that the information contained in this letter and attachments does not alter the conclusions reached in the 10CFR50.92 No Significant Hazards analysis previously submitted.

If you have any questions or require additional information, please contact Mr. Paul Duke at (856) 339-1466.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 11-16-05
(date)

George P. Barnes
George P. Barnes
Site Vice President - Hope Creek

Attachments (2)

C: Mr. S. Collins, Administrator – Region I
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HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
ARTS/MELLLA IMPLEMENTATION

By letter dated June 7, 2004, and in a supplement dated February 18, 2005, PSEG Nuclear LLC (PSEG) requested a revision to the Technical Specifications (TS) for the Hope Creek Generating Station 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). PSEG also proposed to make changes in the methods used to evaluate annulus pressurization (AP) and jet loads resulting from the postulated Recirculation Suction Line Break (RSLB).

In a letter dated August 18, 2005, the NRC staff requested additional information concerning the proposed change. PSEG provided responses to three of the NRC staff's questions in a letter dated September 23, 2005. PSEG's responses to the remaining questions are provided below.

1. It was stated in the submittal for fuel-dependent analysis (NEDC-33066P, Rev. 2) that in general, the limiting anticipated operational occurrences (AOOs) minimum critical power ratio (MCPR) calculation and the reactor vessel overpressure protection analysis are fuel dependent. These analyses are based on, or are applicable to, the reference loading pattern for Cycle 13. Hope Creek is currently operating with approximately 600 irradiated (for one to two previous cycles) non-GE14 fuel assemblies (SVEA-96+) in the core and has loaded 164 fresh GE14 fuel [assemblies] in the Cycle 13 core. For the fuel-dependent evaluations of the limiting AOOs, the licensee's analyses indicate that the original licensed MCPR (OLMCPR) for operation in the MELLLA region remains bounded by the generic power and flow-dependent limits. The analyses results also indicate that performance in the MELLLA region is within allowable design limits for over-pressure protection, loss-of-coolant accidents (LOCAs), and Anticipated Transient Without Scram (ATWS) licensing criteria.

   Describe, in detail, how different fuel designs (for the case of the current loading and for planned future loadings) can be combined to perform a mixed-core analysis and still be able to obtain a bounding fuel-dependent analyses that demonstrates, at MELLLA conditions, that the core and fuel performance will satisfy all safety and regulatory requirements. As a part of this discussion, the response should include:

   a. A discussion of the neutronic and thermal-hydraulic compatibility of the different vendor's fuel loaded into the core.

b. A discussion on which fuel-type (SVEA-96+ or GE14) is more limiting from the standpoint of AOOs, over-pressure protection, LOCA, ATWS, and the thermal-hydraulic stability, including what makes it the limiting fuel-type.

PSEG Response:
Response to Part a
The SVEA-96+ fuel type is treated as a unique fuel design in the standard GE design and licensing process. Sufficient base data about the SVEA-96+ was obtained to model the fuel type's neutronic and thermal-hydraulic characteristics. ISCOR (in Reference 1-1, GESTAR-II) is used to establish the thermal-hydraulic compatibility of the SVEA-96+ fuel loaded into a core with GE14 fuel. ISCOR performs steady state thermal-hydraulic analyses of a nuclear reactor core. ISCOR is the code that implements the NRC approved methodology for performing steady state thermal-hydraulic evaluations. Inputs required for the code include reactor core power level and distribution, inlet flow conditions, reactor core operating pressure and a hydraulic description of the reactor fuel bundles. The GE14 and SVEA-96+ bundles have a separate and unique fuel design specific hydraulic description. The code calculates the core flow distribution and core pressure drop for a given inlet core flow. The code considers the pressure drop and flow in the reactor core only. Detailed modeling of the bypass region, leakage flow paths, water rod hydraulics, and calculated contributions due to friction, local losses, elevation and acceleration is included. Thermal performance calculations are carried out using the GEXL14 critical quality-boiling length correlation for GE14 fuel (Reference 1-2). Thermal performance calculations for SVEA-96+ fuel are carried out using the GEXL80 critical quality-boiling length correlation (Reference 1-3).

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 compatible. Examples of the thermal-hydraulic compatibility for various combinations of the SVEA-96+ and GE14 fuel are provided in the tables below. The examples provided below are for power / flow conditions beyond the range of the MELLLA operating domain being requested in NEDC-33066P, Rev. 2; however, the acceptable thermal-hydraulic compatibility indicated by the results is applicable to the MELLLA operating domain defined in NEDC-33066P, Rev. 2. Furthermore, the examples illustrate that even at conditions beyond the MELLLA operating domain described in NEDC-33066P, Rev. 2, thermal-hydraulic compatibility would be maintained.

Core Performance – Rated Power, Increase Core Flow
(3952 MWth & 105.0 Mlb/hr)

Core Composition		Core Quantities			Hot Bundle[1] Active Flow, klb/hr	
SVEA-96+	GE14	Pressure Drop, psi	Bypass Flow, %[2]	Water Rod WC+WW Flow, %[2]	SVEA-96+	GE14
[[						
						]]

Core Performance – Rated Power, MELLLA Rod Line
(3952 MWth & 99.0 Mlb/hr)

Core Composition		Core Quantities			Hot Bundle[1] Active Flow, klb/hr	
SVEA-96+	GE14	Pressure Drop, psi	Bypass Flow, %[2]	Water Rod WC+WW Flow, %[2]	SVEA-96+	GE14
[[						
						]]

Notes:
[1] Hot Channel Power Peaking Factor = 1.56.
[2] Total flow as percentage of core flow.

The neutronic characteristics for the Hope Creek Generating Station (HCGS) application have been determined using the TGBLA06/PANAC11 methodology (NRC approved per GESTAR II, Amendment 26). As the lattice physics code, TGBLA explicitly models the two-dimensional flux distribution for each unique geometric and compositional arrangement of fuel and burnable poison in the core loading. These multi-group fluxes are generated at multiple historical and instantaneous conditions for synthesizing nodal constants for the three-dimensional core simulator, PANACEA. PANACEA is subsequently used for detailed three-dimensional design and operational calculations of BWR neutron flux and power distributions and thermal performance as a function of control rod position, refueling pattern, coolant flow, reactor pressure, and other operational and design variables. All fuel types are modeled in detail with each fuel type's respective thermal-hydraulic characteristics as well as their respective GEXL correlations specified. Thus, whether the core has multiple reloads of SVEA-96+ and smaller fractions of GE14 or the converse, each and every fuel type is specifically modeled and analyzed in its defined core location.

The compatibility between fuel types and demonstration of the modeling capability for mixed core situations has been validated by a multi cycle benchmark of HCGS SVEA-96+ transition cores. This definitive benchmark comparison of key parameters such as hot and cold eigenvalue, power distribution limits and TIPS validated that the SVEA-96+ fuel design was compatible with the integrated TGBLA06/PANAC11 methodology. As an example of this comparison, the following table provides a summary of the power distribution comparisons conducted from Cycle 9 (all GE fuel) to Cycle 12 (third batch SVEA-96+).

TIP Comparison Summary

Cycle	Statistic	Bundle RMS	Nodal RMS	Axial RMS
9	Avg	[[		
9	Stdev			
10	Avg			
10	Stdev			
11	Avg			
11	Stdev			
12	Avg			
12	Stdev			
All	Avg			
All	Stdev			]]

The average over the cycles 9-12 for bundle RMS is [[ ]] and for nodal RMS is [[ ]]. The standard deviations are [[ ]] and [[ ]], respectively. These numbers are reasonable for a gamma TIP plant. These TIP comparisons for HCGS demonstrate that the TGBLA06/PANAC11 methodology is capable of predicting the mixed core environment well. Additionally, both the summary statistics for all cycles and trends within cycles do not present a departure from the GE experience base.

In addition to multi cycle benchmark comparisons, lattice specific benchmarking for the SVEA-96+ design was conducted. At one or more specific combinations of lattice design, exposure, void content, historical void content, boron inventory, and control state, the lattice reactivity and local pin fission density was compared between TGBLA and the MCNP Monte Carlo neutron transport program. The result of these investigations was that SVEA-96+ is modeled acceptably by TGBLA. Combining this lattice specific benchmark scope with the integrated TGBLA/PANACEA HCGS specific multi cycle benchmark demonstrates a compatible process relative to the ability of the GE methods to model the SVEA-96+ fuel in addition to the existing capability to model the GE14 fuel.

Cycle exposure accounting, which examines hot and cold eigenvalue trending and comparison of simulated thermal limits to monitored thermal limits, continues to be performed for HCGS. This provides on-going confirmation that the mixed core of GE14 and SVEA-96+ continues to be compatible from a neutronic perspective.

Response to Part b

The determination of the limiting bundle for a specific analysis condition involves several factors that depend on power and exposure distributions in the core. In the design process the bundle and core designs are developed to meet the energy requirement and satisfy the required thermal margins for each fuel type. The process represents each fuel type specifically (as described in Response 1a) so that the cycle dependent and independent evaluations determine appropriate operating limits for each fuel type or the results of the evaluations demonstrate acceptable performance relative to acceptance criteria for each fuel type. A determination of which fuel type is limiting is not required to establish acceptable operating limits or to demonstrate acceptable performance relative to acceptance criteria. Either fuel type may be limiting for a specific analysis condition.

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 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 as described in Response 1a). These transient analyses also consider the ARTS/MELLLA operating domain for establishing initial conditions for the transient initiation. The plant transient model calculates the time dependent plant and core response to the prescribed transient using a 1-D axial representation of the core (Reference 1-4). The core nuclear characteristics for the fuel in the core are collapsed into a 1-D representation using appropriate spatial weighting factors for each radial element of the actual 3-D core. The core thermal-hydraulic characteristics are represented in the 1-D model by an average thermal-hydraulic channel. The average channel used to represent the core in the 1-D model is specified so as to preserve the core average thermal-hydraulics characteristics. Based on this 1-D representation of the core, time dependent parameters are calculated for the AOO being analyzed. These time dependent 1-D parameters, [[

]], are then used to determine the input to hot channel transient models. A hot channel transient model is developed for each fuel type using each fuel type's thermal-hydraulic characteristics. [[

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The same model described above for the rated condition AOO analyses is also utilized for the offrated analyses. [[

]] The AOO analyses also show compliance to the fuel rod thermal-mechanical limits, 1% cladding strain and fuel centerline melt. The GE14 and SVEA-96+ fuel rod thermal-mechanical overpower limits for AOO analyses have been established for application at HCGS including consideration of the ARTS/MELLLA operating domain. Relative to the SVEA-96+ fuel rod thermal-mechanical characteristics, sufficient information was supplied by HCGS to enable the application of the GE methodology to the SVEA-96+ fuel design.

The overpressure protection analysis is a core wide event response analysis that is analyzed using the same methods as described above for the AOO analyses with the exception that the hot channel calculation is eliminated since a ΔCPR result is not required. Note, the overpressure protection analysis is performed each reload cycle and uses the design nuclear and hydraulic characteristics of the reload core (see Response 1a).

The LOCA analysis is performed for each fuel type in a bounding analysis to be independent of a specific fuel cycle. The LOCA analysis is validated each reload cycle by confirming that key inputs to the SAFER analyses are unchanged and are still valid for the upcoming cycle. The SAFER/GESTR-LOCA methodology (Reference 1-5) assumes an equilibrium core loading for each fuel type. This approach is acceptable because of the channeled configuration of BWR fuel assemblies. There is no channel-to-channel cross flow inside the core and the only issue of hydraulic compatibility of the various bundle types in a core is the bundle inlet flow rate variation. As described in Response 1a, based on the detailed modeling of each fuel design, the key thermal-hydraulic performance parameters were validated for the mixed core to be compatible. As a result, there is no significant difference in the hydraulic response for a mixed core as compared to an equilibrium core. The SAFER analysis is insensitive to mixed cores. The PCT is determined by hot channel response. The hot channel has the fuel placed on the LHGR limits while the bundle power is based on MCPR limits selected to be lower than those expected during operation. The hot bundle hydraulics are driven by the overall core pressure drop. This basic premise is valid because no channel-to-channel interaction occurs during a LOCA.

The first peak PCT is primarily influenced by the timing of boiling transition at the various elevations in the bundle. The boiling transition in the bundle is governed by the core flow coastdown characteristics and the bundle power level. The core flow coastdown is a core-wide phenomenon determined by the initial core flow and the recirculation pump coastdown, neither of which are dependent on the fuel type. The bundle power also affects the boiling transition time; a higher power bundle will experience an earlier and potentially deeper boiling transition. Because of the channeled configuration of BWR fuel assemblies, there is no channel-to-channel cross flow inside the core. The boiling transition in one bundle will not affect the other bundles in the core. The second peak PCT is primarily influenced by bundle flooding from the bottom. This is a low flow rate process that is governed by the ECCS system capacity. There is no channel-to-channel interaction during this time. Therefore, the transition from a mixed core to an equilibrium core will not affect the second peak PCT response.

The ATWS event is a low probability event that was analyzed and addressed as a cycle independent event. The ATWS analysis is based on the use of NRC approved methodology (Reference 1-4). Historically, ATWS analyses have used a nominal basis. [[

]] To further substantiate the cycle / core design independent aspects of the ATWS evaluation, the ATWS analysis study in

NEDC-33066P, Rev. 2 considered both an equilibrium core of GE14 and a transition core with a mixture of GE14 and SVEA-96+. The ATWS study used the same modeling approach for the core as was used in the AOO process described above. The ATWS key safety parameters documented in NEDC-33066P, Rev. 2 were all well below the acceptance criteria.

The thermal-hydraulic stability is addressed with cycle specific calculations that take into account mixed core considerations as described in Response 1a. In addition, see the responses to RAIs 2 (Reference 1-6) and 3 (below).

References:

1-1 Global Nuclear Fuel, Licensing Topical Report, "General Electric Standard Application for Reactor Fuel," NEDE–24011–P–A–15, Class III, September 2005.
1-2 GE Nuclear Energy, "GEXL14 Correlation for GE14 Fuel," NEDC-32851P, Revision 2, Class III, September 2001.
1-3 Global Nuclear Fuel, "GEXL80 Correlation for SVEA96+ Fuel," NEDC-33107P-A, Revision 1, Class III, October 2004.
1-4 GE Nuclear Energy, Licensing Topical Report, "Qualification of the One-Dimensional Core Transient Model (ODYN) for Boiling Water Reactors," NEDO-24154-A, Volumes 1 – 3, Class I, February 1, 1986, NEDC-24154P-A, Supplement 1, Volume 4, Class III, February 2000.
1-5 General Electric Company, Licensing Topical Report, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident, Volume III, SAFER/GESTR Application Methodology," NEDE-23785-1-PA, Revision 1, Class III, October 1984.
1-6 PSEG Nuclear letter LR-N05-0448, "Response to Request for Additional Information," dated September 23, 2005

3. It is the NRC staff's understanding that the vendor's methodologies were approved only to apply to their respective fuel designs. Demonstrate that the DIVOM curve is bounding when the core is loaded with different fuel designs, supplied by different fuel vendors.

PSEG Response:
The current fuel designs among the fuel vendors are in general very similar in terms of nuclear design and physical geometry. From a neutronics point of view, the fuel is mostly uranium-enriched, with an appropriate level of plutonium buildup. The key neutronics driver (void reactivity) is comparable between the different vendors' fuels. To illustrate this point, the EOC dynamic void coefficients (DVCs) for the HCGS Cycle 13 mixed core and a similar sized plant have been computed for rated power/rated flow conditions. The EOC DVC is -0.20 \$/%Δvoid for the HCGS Cycle 13 mixed core (which consists of 600 bundles of SVEA-96+ and 164 bundles of GE14) while the DVC for a similar plant with a full core of GE14 is -0.23 \$/%Δvoid.

Both DVCs are consistent with the range specified (-0.13 to –0.23 \$/%Δvoid) in Table B-4 of the Option III LTR (Reference 3-1).

The physical geometry drives the two-phase/single-phase pressure drop and the fuel thermal time constant and these are explicitly modeled for each fuel design in the GE DIVOM methodology.

As discussed in the response to RAI 1, sufficient data about the SVEA-96+ fuel design was obtained to model the fuel types neutronics and thermal-hydraulic characteristics. In addition, a critical power correlation for the SVEA-96+ fuel (GEXL80) has been developed to provide the correct CPR response. This allows GE to use TRACG, which is the GE proprietary version of the Transient Reactor Analysis Code (TRAC), to establish an accurate DIVOM curve. TRACG uses advanced realistic one-dimensional and three-dimensional methods to model the phenomena that are important in evaluating the stability response of BWRs. Realistic analyses performed with TRACG have been used previously to support licensing applications in different areas including Anticipated Operational Occurrence (AOO) and DIVOM stability.

The DIVOM curve represents a relationship between the hot channel oscillation magnitude and the fractional change in Critical Power Ratio. Per the guidance provided in the BWROG Regional DIVOM Guideline, a plant and cycle-specific DIVOM evaluation will be performed at the limiting power/flow condition for every reload at three exposure state points (BOC, MOC or Peak Hot Excess, and EOC). This represents a reasonably limiting DIVOM curve for the stability Option III application.

Consequently, the DIVOM curve developed for the HCGS ARTS/MELLLA application is adequate to be used as an input to the determination of the OPRM system amplitude setpoint since mixed core considerations have been specifically modeled. No change to the approved methodology is required.

Reference:

3-1 GE Nuclear Energy, Licensing Topical Report, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications," NEDO-32465-A, Class I, August 1996.