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JUL 13 2007

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

**SUSQUEHANNA STEAM ELECTRIC STATION
PROPOSED LICENSE AMENDMENT NO. 285
FOR UNIT 1 OPERATING LICENSE NO. NPF-14
AND PROPOSED LICENSE AMENDMENT NO. 253
FOR UNIT 2 OPERATING LICENSE NO. NPF-22
EXTENDED POWER UPRATE APPLICATION
SUPPLEMENT TO INSTRUMENTATION AND CONTROLS
AND BALANCE-OF-PLANT SAFETY SYSTEMS REQUEST
FOR ADDITIONAL INFORMATION RESPONSES**
PLA-6236

**Docket Nos. 50-387
and 50-388**

References:
1) *PPL Letter PLA-6076, B. T. McKinney (PPL) to USNRC, "Proposed License Amendment Numbers 285 For Unit 1 Operating License No. NPF-14 and 253 for Unit 2 Operating License No. NPF-22 Constant Pressure Power Uprate," dated October 11, 2006.*
2) *PPL Letter PLA-6204, B. T. McKinney (PPL) to USNRC, "Proposed License Amendment Numbers 285 For Unit 1 Operating License No. NPF-14 and 253 for Unit 2 Operating License No. NPF-22 Extended Power Uprate Application Re: Instrumentation and Controls Technical Review - Request for Additional Information Responses," dated June 1, 2007.*
3) *PPL Letter PLA-6198, B. T. McKinney (PPL) to USNRC, "Proposed License Amendment Numbers 285 For Unit 1 Operating License No. NPF-14 and 253 for Unit 2 Operating License No. NPF-22 Extended Power Uprate Application Re: Balance-of-Plant Safety Systems - Request for Additional Information Responses," dated May 14, 2007.*

Pursuant to 10 CFR 50.90, PPL Susquehanna LLC (PPL) requested in Reference 1 approval of amendments to the Susquehanna Steam Electric Station (SSES) Unit 1 and Unit 2 Operating Licenses (OLs) and Technical Specifications (TSs) to increase the maximum power level authorized from 3489 Megawatts Thermal (MWt) to 3952 MWt, an approximate 13% increase in thermal power. The proposed Constant Pressure Power Uprate (CPPU) represents an increase of approximately 20% above the Original Licensed Thermal Power (OLTP).

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NRR

The purpose of this letter is to supplement the responses to several Instrument and Controls (I&C) and Balance of Plant (BOP) related NRC Questions that are contained in the Request for Additional Information Responses transmitted to NRC in References 2 and 3 based on the teleconference held with the NRC Staff.

Attachment 1 contains PPL responses to the I&C questions and contains information that is AREVA NP, Inc. and GE-Hitachi Nuclear Energy Americas LLC proprietary. As such, AREVA NP, Inc. and GE-Hitachi Nuclear Energy Americas LLC request that Attachment 1 be withheld from public disclosure in accordance with 10 CFR 2.390 (a)4. The affidavits supporting this request are contained in Attachment 4. Attachment 2 contains a non-proprietary version of Attachment 1. Attachment 3 contains PPL's responses to the BOP questions.

There are no regulatory commitments associated with this submittal.

PPL has reviewed the "No Significant Hazards Consideration" and the "Environmental Consideration" submitted with Reference 1 relative to the Attachments. We have determined that there are no changes required to either of these documents.

If you have any questions or require additional information, please contact Mr. Michael H. Crowthers at (610) 774-7766.

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

Executed on: 7-13-07

B. T. McKinney

B. T. McKinney

Attachment 1: Proprietary Supplemental Response to I&C Information Request
Attachment 2: Non-Proprietary Supplemental Response to I&C Information Request
Attachment 3: Supplemental Response to BOP Request for Additional Information
Attachment 4: AREVA NP, Inc. and GE-Hitachi Nuclear Energy Americas LLC Affidavits

Copy: NRC Region I
Mr. A. J. Blamey, NRC Sr. Resident Inspector
Mr. R. V. Guzman, NRC Sr. Project Manager
Mr. R. R. Janati, DEP/BRP

Attachment 2 to PLA-6236
Non-Proprietary Supplemental Response to I&C Request for Additional Information

The following provides supplemental information to several NRC Questions contained in the Request for Additional Information Responses transmitted to NRC in References 2 and 3.

Reference 2: PLA-6204 Response to NRC Instrumentation and Controls Technical Review Request for Additional Information

Based on a teleconference held with the NRC Staff on June 13, 2007, the following additional information is provided relative to the PPL Responses to NRC Questions 5, 6 and 7 in PLA 6204.

PPL follow-up to NRC Question 5:

In PLA-6204 PPL Response to NRC Question 5, PPL described that the uncertainty increases to 4.3% for calibration intervals of 2000 MWD/MTU due to the longer exposure period. Comparisons between actual plant data for the LPRM calibration current vs. the predicted calibration values over various time intervals have been performed to evaluate the uncertainty change caused by the LPRM calibration interval extension. The comparisons were performed using actual plant calibration data from a BWR/5 and a BWR/6. The BWR/5 is similar to SSES since they both have the same size core and C-lattice geometry with NA300 LPRM detectors, and is therefore applicable to SSES. The increases in the relative standard deviation value were then added to the uncertainty value of 3.4% related to 1000 MWD/MT exposure interval (Reference 1), and compared to the value of 4.3% which is used in plant transient analysis and reload analysis. Results showed that LPRM uncertainties for 2000 MWd/MTU (with an allowance for 25% to 2500 MWd/MTU) are still bound by the value of 4.3%.

The predicted calibration currents are calculated by the following equation:

$$I_n = I_{n-1} e^{\lambda E}$$

where I_n is the predicted calibration current

I_{n-1} is the plant calibration current from last LPRM calibration

λ is the effective exposure decay factor for the LPRM detector (1/snvt)

E is the accumulated LPRM detector exposure (snvt) since last LPRM calibration time.

A range of calibration intervals are grouped together to quantify the uncertainty at a particular interval. For example:

1000 MWd/MTU is associated with all data between [ ] MWd/MTU

2500 MWd/MTU is associated with all data between [ ] MWd/MTU

The MCPR safety limit calculation is affected by the change in the bundle power distribution uncertainties caused by the LPRM calibration interval extension. The following discussion provides specific details as to how the increased LPRM uncertainties affect the bundle power distribution uncertainties in the MCPR safety limit calculation.

The generic uncertainty analysis which quantifies the uncertainty associated with BWR core monitoring using the POWERPLEX-III Core Monitoring Software System is presented in Reference 2. The interrelation among the individual uncertainty components and the radial bundle power uncertainty, $\delta_{P^n_{ij}}$, is described here in Figure 1. It employs circles to designate uncertainty components evaluated from basic measured or calculated data. It employs rectangles to designate uncertainty components whose value depends on the methodology (propagation).

The uncertainty components in Figure 1 which require re-evaluation in order to investigate the impact of extension of the LPRM detector calibration interval from 1000 MWD/MT to 2000 MWD/MT on the generic uncertainty quantities are the [ ], and the [ ].

Equation 9-13 in Reference 0, repeated below, defines $\delta_{D_{ij}}$,

[

]

]

Where:

$\delta_{TIP_{ij}}$ = Relative standard deviation in the (20 axial data point average) TIP distribution.

$\frac{\delta_{LPRM}}{4}$ = Relative standard deviation in the average of the four LPRM responses at a TIP location.

$\delta_{S_{ij}}$ = Relative standard deviation due to the synthesis procedure.

Thus, the only uncertainty change associated with the LPRM calibration interval extension that contributes to the bundle power distribution uncertainties calculation and MCPR safety limit calculation is [ ]

References:

1. NEDO-20340, "process Computer Performance Evaluation Accuracy," June 1974.

2. EMF-2158 (P)(A) Revision 0, "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2," October 1999.

Legend:

$\delta_{P_{ij}^{n}}$ = Radial Bundle Power Uncertainty

$\delta_{B_{ij}}$ = Power Distribution Calculation Uncertainty

$\delta_{D_{ij}}$ = Synthesized TIP Distribution Uncertainty

$\delta_{T_{ij}}$ = TIP Distribution Calculation Uncertainty

P_{ij} = Correlation Coefficient

$\delta_{TIP_{ij}}$ = TIP Distribution Measurement Uncertainty

δ_{LPRM} = LPRM Measurement Uncertainty

$\delta_{S_{ij}}$ = Synthesis Procedure Uncertainty

$\delta_{T_i^l}$ = Relative Standard Deviation on the Differences between Measured and Calculated TIP Distributions

Figure 1 Relationship among Uncertainty Components

PPL follow-up to NRC Question 6:

In response to NRC Question 6, PPL provided the explanation and the equations for calculating the differential pressure. In addition, PPL provided the General Electric calculation for the main steam line high flow function in Attachment 4. This calculation is based on an assumed Maintenance and Testing Equipment (M&TE) accuracy. The actual accuracy of the M&TE equipment used for calibration will be equal to or better than the M&TE accuracy assumed in the calculation. The PPL standard for M&TE accuracy is to use M&TE instruments that are at least 4 times more accurate than the instrument being calibrated. Thus, for the Main Steam Line Differential Pressure Indicating Switches with an accuracy of ± 3%, the M&TE would have an accuracy of ±0.75% or better. Typically, for these instruments, a Heise gauge that has an accuracy of ± 0.1%FS is used. Thus for a 200 psi FS range, the accuracy would be ± 0.2 psi, well within the ± 2.2 psi assumed by GE in the setpoint calculation.

PPL follow-up to NRC Questions 5 and 7: (Setpoint Methodology)

Per the discussion with NRC on 6/13/2007, the "restrictions" referred to in the question are specifically those listed on pages 4 and 5 of the SER for NEDC-33004P-A. Based on the following excerpt from page 4 of the NEDC-33004-P-A SER, the intent of the listed restriction is to reduce the complexity of CPPU submittals by focusing the review on the changes required for the CPPU alone.

> *"These factors have been incorporated into the overall approach to simplify the required plant specific documentation while maintaining a systematic licensing and safety evaluation process. Further, the focus of the evaluation has been placed on the safety evaluations required for CPPU alone, with out changes to maximum core flow,... to allow for a comprehensive but more streamlined NRC staff review process."*

None of the restrictions specified in the NEDC-33004-P-A SER are related to the use of the GE Setpoint Methodology NEDC-31336P-A. The APRM flow biased simulated thermal power based scram setpoints are determined using the GE Setpoint Methodology NEDC-31336P-A. Use of this methodology for these setpoints was reviewed and accepted by the NRC in the NRC Safety Evaluation Report entitled "Susquehanna Steam Electric Station, Units 1 and 2 - Issuance of Amendment Re: Average Power Range Monitor/Rod Block Monitor/Technical Specifications/maximum extended load Line Limit Analysis (MELLLA) implementation (TAC NOS. MC09040 and MC9041" issued March 23, 2007.

Both SSES Units contain full cores of ATRIUM-10 fuel. The SSES CPPU submittal conforms to the previous discussed restriction since [
] Fuel related safety analyses for ATRIUM-10 fuel have been performed by the fuel vendor unless specifically noted.

The NEDC-33004P-A simplified calculation methodology can be used to determine the APRM flow biased simulated thermal power based scram setpoints because [

]

Therefore, based on all of the above reasons, the use of the GE simplified setpoint methodology with non-GE fuel at SSES is justified.

Attachment 3 to PLA-6236
Supplemental Response to BOP Request for Additional Information

Reference 3: PLA-6198 Response to NRC Balance of Plant Technical Review Additional Information

Based on a teleconference held with the NRC Staff on June 14, 2007, the following additional information is provided relative to the PPL Responses to NRC Questions 1 and 4 from PLA 6198. Also, NRC Staff provided an additional request for information beyond what was requested and responded to in PLA-6198. This question and the PPL response is provided herein and is identified as new NRC Question 5.

NRC Question 1:

The response does not indicate that an actual cycle specific analyses is currently being performed or will be performed after CPPU to ensure pool temperature limits are not exceeded. Attachment 2 to Matrix 5 of Review Standard RS-001, states in Section 3.1 that: "The licensee demonstrates adequate SFP cooling capacity by either performing a bounding evaluation or committing to a method of performing outage-specific evaluations." The licensee CPPU analysis, based on its response to Question 1a, 1b, and 1c, is not consistent with what would be considered a bounding analysis and they have not provided information to show that the CPPU analysis is consistent with the plant's current licensing basis SFP cooling evaluation. In addition, the licensee has not committed to perform cycle specific evaluations to demonstrate adequate SFP cooling capability. Based on a review of the material that was provided in PLA-6198, the licensee has not fully satisfied either option.

PPL Response 1:

The fundamental SSES licensing requirements related to the fuel pool cooling system are stated in Section 6.3.1 of the PUSAR (Attachment 4 Reference PLA-6076). These requirements are: 1) maintain the spent fuel pool (SFP) bulk temperature below 125°F; and, 2) maintain a "time to boil" of at least 25 hours when a Seismic Category 1, Class 1E cooling system is not assisting in fuel pool cooling. These licensing requirements, which are more restrictive than those in SRP Section 9.1.3, are not being changed for CPPU.

In order to assure that these licensing requirements are satisfied, outage-specific calculations are performed to ensure that RHR is not removed from service until the "time to boil" exceeds 25 hours, and the SFP bulk temperature does not exceed the administrative limit of 115°F (vs. the licensing limit of 125°F). To ensure that the administrative limit can be maintained, these calculations assume makeup and service water temperatures which are slightly higher than the actual temperatures expected during the outage.

These calculations are mandated by plant procedures, which are subject to the provisions of 10 CFR 50.59. Thus, any significant change to the current program which assures these requirements are satisfied would likewise be subject to that process. While the SSES FSAR does not specifically state that outage-specific analyses will be performed, plant procedures are in place to satisfy the mandated licensing commitments.

For current plant operation, one RHR system pump needs to be operated in the shutdown cooling (SDC) mode for approximately 4½ days after all the rods have been inserted to meet the 25 hour "time to boil" licensing requirement. The calculation of "time to boil" is based on the current fuel operating cycle (plus the existing spent fuel) decay heat and is independent of cooling water temperature. At CPPU conditions, one RHR pump in SDC mode will need to be in service for approximately 5½ days to meet 25 hour requirement.

The SFP analysis contained in the PUSAR (Attachment 4 Reference PLA-6076) states that a bounding analysis was performed. The cooling water temperatures used in the CPPU analysis conservatively bound the actual cooling water temperatures experienced during a typical spring refueling outage. The CPPU analysis assumed a river makeup temperature of 75°F and a service water temperature of 95°F. A typical spring refueling outage experiences a river makeup temperature of approximately 40°F and a service water temperature of approximately 75°F. Thus, the temperatures assumed in the CPPU analysis bound those seen during a typical spring refueling outage.

NRC Question 4:

NRC Question 4 concerns the licensee's proposed modification to the ultimate heat sink. The question asked the licensee to explain how modification testing will be used to verify that the actual height of spray is consistent with the analytical model spray height results. In its response, the licensee states, "testing does not need to be used to verify the actual height of spray from the most limiting nozzle is consistent with the analytical spray height results." The staff is not in agreement with the licensee's response and believes that the licensee needs to commit to a test to verify the spray height achieved is consistent with those used in the analytical model. Because the spray height is relied upon for assuring adequate decay heat removal, the NRC Staff has determined that adequate testing for demonstrating the assumed capability is required. This is considered to be an open item pending the establishment of testing for the post modify spray pond that will verify the analytical model assumptions for spray height is correct.

PPL Response 4:

The correlation between spray height and nozzle pressure is based on nozzle vendor data and bench test data points that were obtained when the Susquehanna nozzles were originally purchased. Most of the nozzles have been in service since approximately 1981. Various inspections and maintenance activities have not identified any flow-induced erosion or degradation of the nozzles since installation of the nozzles. Therefore, the correlation between spray height and nozzle pressure remains valid for the UHS analysis performed for CPPU. An additional inspection was performed July 9, 2007 to provide additional confirmation that erosion or degradation of the nozzles has not taken place. A sampling of spray nozzles on each of the four spray networks was inspected. The inspection included four or five nozzles on each of the four networks. No erosion or degradation of the bronze nozzles was noted. The exit orifices, which are critical for nozzle performance, were clean and smooth and not eroded in all cases.

As a result, testing to demonstrate that the spray nozzle spray height is consistent with the analytical spray height results is not required.

NRC Question 5:

The Feedwater/Condensate pump trip test that is planned to be performed during CPPU power ascension testing is discussed on Page 12 of a Attachment 8 of the CPPU submittal. In the discussion, it is indicated that a condensate pump trip will be conducted at the 3733 MWt power level, after the first stage of the uprate is implemented. The test at the full uprated power after the implementation of the second stage of the uprate is only agreed to conditionally. The licensee states that only if the results of the 3733 MWt test are not sufficient to reasonably confirm the analysis model used for the 3733 MWt step, will the condensate pump trip test be repeated at 3952 MWt.

The staff's concern is that for the uprated plant, there may not be adequate available NPSH after a trip of a feedwater or condensate pump(s) to prevent other feedwater pumps from tripping on low suction pressure, and causing a total loss of feedwater event. The licensee proposed condensate trip test would be a means of demonstrating the system capability to sustain a condensate, and feedwater pump trip without a loss of feed. The staff requests licensee to: 1) confirm the condensate pump trip bounds the feedwater pump trip with respect to the most limiting feedwater pump NSPH response in terms of NPSH, and 2) provide the acceptance criteria and basis for determining if the test must be repeated at 3952 MWt, including how the available margin must compare to the required margin if the test is not repeated.

PPL Response 5:

The condensate pump trip does bound the feedwater pump trip in terms of NPSH. The minimum Reactor Feedwater Pump (RFP) suction pressure during a Feedwater Pump Trip from CPPU conditions is 399 psig and the minimum RFP suction pressure during a condensate pump trip is 248 psig. The trip setpoint corresponding to the required net positive suction head ($NPSH_r$) is 285 psig.

The acceptance criteria is as follows:

Level 1 Criteria:

The trip of one condensate pump shall not cause the trip of all three feedwater pumps.

Level 2 Criteria:

The trip of one condensate pump shall not cause the trip of more than one feedwater pump. A recirculation runback shall occur upon the trip of a condensate pump. For the 3733 MWt test only, the margin to the RFP suction pressure trip shall not be less than 10 psi.

For CPPU, the time delays on the RFP low suction pressure trip have been changed as follows:

	A RFP	B RFP	C RFP
Current	5 ±0.5 sec	10 ±1.0 sec	15 ±1.5 sec
CPPU	5 ±0.5 sec	15 ±1.5 sec	30 ±3.0 sec

When a condensate pump trip occurs, the RFP suction pressure will decrease to 248 psig. The trip setpoint corresponding to the required net positive suction head ($NPSH_r$) is 285 psig. The decrease of the suction pressure below the trip setpoint will start the timers in the Reactor Feedwater Pump Turbine (RFPT) trip circuits on all three RFPs. Due to the tripping of the condensate pump, the recirculation system receives a runback signal to reduce recirculation pump speed to the #2 limiter setpoint. The RFPTs will increase speed towards their High Speed Stop setpoints.

At 5 ±0.5 sec, the 'A' RFPT will trip, reducing feedwater flow. When the RFP trips, the drop in discharge pressure rapidly closes the discharge check valve causing a rapid reduction in feedwater flow that initiates suction pressure recovery. Approximately 3 – 5 seconds after the 'A' RFP trip, the suction pressure recovers to 388 psig, resetting the trip logic on the 'B' and 'C' RFPTs when the pressure increases above 315 psig. The suction pressures in this scenario are based on the RFPTs being at their high speed stops, thus producing the maximum flow and minimum suction pressure. Reactor power will be decreasing due to the recirculation pump speed runback, thus reducing the need for feedwater flow, so the RFPT speed will begin to decrease, further increasing RFP suction pressure.

In the unlikely event that the suction pressure does not recover in time to prevent the trip of the second RFPT, once the second RFPT trips, the suction pressure will recover to 546 psig, however, the flow would not be sufficient to maintain water level and the reactor would SCRAM on low water level. The remaining RFP would be sufficient to maintain water level after the SCRAM.

It should be noted that minor uncertainties in the suction pressure calculation do not affect the scenario because the pressures are significantly below the trip setting of 285 psig after the condensate pump trips and significantly above the trip reset after the RFP trips. The major effect in the scenario is caused by uncertainties in the RFPT suction pressure trip timer settings and the pressure recovery time. The pump manufacturer states that pressure recovery occurs between 3 and 5 seconds. Plant data during a RFPT trip indicates pressure recovery occurs between 1 and 3 seconds. The time delays between RFP trips were chosen to ensure feedwater would remain available for reactor makeup after a condensate pump trip. In the worst case, the first pump trip occurs at 5.5 seconds and pressure recovery takes 5 seconds. That leaves three seconds for the timers to reset before the second pump trips at the minimum time of 13.5 seconds. The time delays were verified by the pump manufacturer to be acceptable.

Performance of the condensate pump trip at the 3733 MWt power level will allow validation of the RFP suction pressure / flow relationship. This relationship will then be used to assess performance at 3952 MWt and determine if adequate NPSH will be available after a trip of a condensate pump to prevent other feedwater pumps from tripping on low suction pressure, causing a total loss of feedwater event.

The reactor is not expected to SCRAM on low water level with a condensate pump trip from 3733 MWt. Depending on the initial conditions of the reactor (power/flow), a reactor SCRAM on low water level at 3952 MWt is more likely.

The modifications to the condensate / feedwater system that will be made for the power increase to 3952 MWt that will not be in place during the 3733 MWt test are:

- Reactor Feedwater Pump Turbine Upgrade (Increases RFPT maximum speed from 5200 rpm to 5585 rpm)
- Installation of one additional condensate filter
- Installation of one additional condensate demineralizer

The SSES condensate / feedwater hydraulic model is a steady-state detailed multi-node model of the system that solves the Darcy-Weisbach formula for each node and uses pump affinity laws to calculate the $NPSH_r$ for the higher RFP speeds and flows required for CPPU. The model has parallel paths where appropriate (i.e. four condensate pumps, three feedwater pumps, seven filters and eight demineralizers). The model uses a maximum speed of 5200 rpm for the 3733 MWt cases and 5585 rpm for the 3952 MWt cases and the appropriate flow rates and temperatures. There are no other effects from the Reactor Feedwater Pump Turbine Upgrade modification. The additional condensate filter and demineralizer are included in the model for the 3952 MWt cases and are not included for the 3733 MWt cases. The additional condensate filter and demineralizer are in parallel with the other filters and demineralizers, so the effect on the overall pressure drop of the additional components, piping and fittings is offset by the reduction in flow through each of the parallel components. The net effect is a slight (<1 psi) increase in overall pressure drop in the condensate / feedwater system. Because the model has been benchmarked to plant data by adjusting friction factors and loss coefficients as appropriate and there is a negligible effect of the additional condensate filter and demineralizer, it will be valid for use at CPPU conditions.

The only time the plant will be at 3952 MWt is in the summer time (July/August) on a hot day. In hot days in the summer, the PJM grid tends to be heavily loaded. A SCRAM of one SSES unit at that time would put a significant transient on the PJM grid. Given the potential consequences and since the bounding analysis of a total loss of feedwater flow is not a limiting transient from a thermal margin perspective; PPL has determined that performance of the test at 3952 MWt is not warranted providing the test at 3733 MWt is satisfactory.

Attachment 4 to PLA-6236
AREVA NP, Inc. and
GE-Hitachi Nuclear Energy Americas LLC
Affidavits

GE-Hitachi Nuclear Energy Americas LLC

AFFIDAVIT

I, James F. Harrison, state as follows:

(1) I am Project Manager, Fuel Licensing, Regulatory Affairs, GE-Hitachi Nuclear Energy Americas LLC ("GEH"), have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in GEH letter, GE-SSES-AEP-330, *GEH Proprietary Review of PPL Letter, PLA-6236*, dated July 5, 2007. The proprietary information, contained in Enclosure 1 entitled, *GEH Proprietary Review of PPL Letter, PLA-6236,* is delineated by a [[dotted underline inside double square brackets.{3}]]. In each case, the superscript notation {3} refers to Paragraph (3) of this affidavit, which provides the basis for the proprietary determination.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Executed on this 5th day of July 2007.

James F. Harrison

James F. Harrison
Project Manager, Fuel Licensing, Regulatory Affairs
GE-Hitachi Nuclear Energy Americas LLC

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Gayle F. Elliott. I am Manager, Product Licensing, for AREVA NP Inc. and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA NP to determine whether certain AREVA NP information is proprietary. I am familiar with the policies established by AREVA NP to ensure the proper application of these criteria.

3. I am familiar with the AREVA NP information contained in the PPL letter PLA-6236, "Susquehanna Steam Electric Station Proposed License Amendment No. 285 for Unit 1 Operating License No. NPF-14 and Proposed License Amendment No. 253 for Unit 2 Operating License No. NPF-22 Extended Power Uprate Application Supplement to Instrumentation and Controls and Balance-of-Plant Safety Systems Request for Additional Information Responses," dated June 2007, and referred to herein as "Document." Information contained in this Document has been classified by AREVA NP as proprietary in accordance with the policies established by AREVA NP for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA NP and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by AREVA NP to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA NP's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA NP.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA NP in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA NP, would be helpful to competitors to AREVA NP, and would likely cause substantial harm to the competitive position of AREVA NP.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(b) and 6(c) above.

7. In accordance with AREVA NP's policies governing the protection and control of information, proprietary information contained in this Document have been made available,

on a limited basis, to others outside AREVA NP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA NP policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 10th

day of July, 2007.

Sherry L. McFaden
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 10/31/10