Progress Energy

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10 CFR 50.90

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

Subject: Brunswick Steam Electric Plant, Unit Nos. 1 and 2
Renewed Facility Operating License Nos. DPR-71 and DPR-62
Docket Nos. 50-325 and 50-324
Request for License Amendments to Revise Local Power Range Monitor Calibration Frequency

Ladies and Gentlemen:

In accordance with the Code of Federal Regulations, Title 10, Part 50.90, Carolina Power & Light Company (CP&L), now doing business as Progress Energy Carolinas, Inc., is requesting a revision to the Technical Specifications (TSs) for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2. The proposed license amendments revise Technical Specification (TS) 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Surveillance Requirement (SR) 3.3.1.1.8 to increase the frequency interval between Local Power Range Monitor (LPRM) calibrations from 1100 megawatt-days per metric ton (MWD/T) average core exposure (i.e., equivalent to approximately 907 effective full power hours (EFPH)) to 2000 EFPH. An evaluation of the proposed license amendments is provided in Enclosure 1.

CP&L has evaluated the proposed change in accordance with 10 CFR 50.91(a)(1), using the criteria in 10 CFR 50.92(c), and determined that this change involves no significant hazards considerations.

In accordance with 10 CFR 50.91(b), CP&L is providing, a copy of the proposed license amendments to the designated representative for the State of North Carolina.

CP&L requests approval of the proposed amendments by February 26, 2010. Once approved, the Unit 1 amendment shall be implemented prior to start-up from the 2010 Unit 1 refueling outage and the Unit 2 amendment shall be implemented prior to start-up from the 2011 Unit 2 refueling outage.

There are no regulatory commitments associated with the proposed amendments. Please refer any questions regarding this submittal to Ms. Annette Pope, Supervisor – Licensing/Regulatory Programs, at (910) 457-2184.

P.O. Box 10429
Southport, NC 28461

T> 910.457.3698

A001
NRR

I declare, under penalty of perjury, that the foregoing is true and correct. Executed on August 18, 2009.

Sincerely,

Benjamin C. Waldrep

WRM/wrm

Enclosures:

1. Evaluation of License Amendment Request
2. Marked-up Technical Specification Pages - Unit 1
3. Typed Technical Specification Page - Unit 1
4. Typed Technical Specification Page - Unit 2
5. Marked-up Technical Specification Bases Page - Unit 1 (For information only)
6. Calculation 0B21-1305, *Core Monitoring LPRM Uncertainty and Sensitivity Decay*

cc (with enclosures):

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Evaluation of Proposed License Amendment Request

Subject: Request for License Amendments to Revise Local Power Range Monitor Calibration Frequency

1.0 Description

This letter is a request by Carolina Power & Light Company (CP&L), now doing business as Progress Energy Carolinas, Inc., to amend the Technical Specifications (TS) for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2. The proposed license amendments revise Technical Specification (TS) 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Surveillance Requirement (SR) 3.3.1.1.8, to increase the frequency interval between Local Power Range Monitor (LPRM) calibrations from 1100 megawatt-days per metric ton (MWD/T) average core exposure (i.e., equivalent to approximately 907 effective full power hours (EFPH)) to 2000 EFPH.

2.0 Proposed Change

The purpose of the proposed change is to revise the TS SRs for periodic calibration of the LPRMs. The current requirement is specified by SR 3.3.1.1.8. This surveillance requirement currently specifies that the LPRMs be calibrated at a frequency of every 1100 MWD/T average core exposure. The proposed change will revise the frequency of surveillance to every 2000 EFPH and will read as follows:

Current

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.8	Calibrate the local power range monitors.	1100 MWD/T average core exposure

Proposed

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.8	Calibrate the local power range monitors.	2000 effective full power hours

For convenience, Enclosure 2 contains a marked-up version of the Unit 1 TSs showing the proposed changes. Since SR 3.3.1.1.8 for Unit 1 and Unit 2 are identical, only the

mark-up for Unit 1 is provided. Enclosures 3 and 4 provide typed versions of the Unit 1 and Unit 2 TSs, respectively. These typed TS pages are to be used for issuance of the proposed amendment.

In addition, in support of this proposed TS change, the associated TS Bases Section 3.3.1.1.8 will be revised to reflect the change in the LPRM calibration frequency from 1100 MWD/T average core exposure to 2000 EFPH. The TS Bases changes are provided in Enclosure 5, for information only, and do not require NRC approval.

3.0 Technical Evaluation

Background

The LPRM system consists of fission chamber detectors, signal conditioning equipment, display and alarm equipment, associated power supplies, cabling, and trip functions. The LPRM system consists of 31 LPRM detector strings distributed radially throughout the core. Each detector string contains four fission detectors located at four different fixed axial heights. Each fission chamber produces a current that is coupled with the LPRM signal conditioning equipment to provide the desired scale indications. The chambers are vertically spaced in the LPRM detector dry tube assemblies to monitor four horizontal planes of the core, complementing the radial coverage given by the arrangement of the LPRM detector dry tube assemblies across the core.

Each LPRM dry tube assembly also contains a calibration tube for a Traversing In-core Probe (TIP). The TIP movable fission detectors are periodically traversed to provide a continuous axial flux profile at each LPRM string location. From these gamma flux profiles, thermal neutron flux profiles are calculated. Appropriate Gain Adjustment Factors (GAFs) are determined for each LPRM detector based on this information. These GAF values are then applied to LPRM signals during the calibration of the 124 fixed LPRM fission detectors. These calibrations compensate for changes in detector sensitivity resulting from the depletion of fissile material lining the individual LPRM fission chambers. LPRM calibrations are performed while the reactor is operating at power due to the limited sensitivity of the LPRM detectors.

The LPRM system provides outputs to the Average Power Range Monitor (APRM) system, the Rod Block Monitor (RBM) system, the Oscillation Power Range Monitor (OPRM) system, the core monitoring system, and the plant process computer. The APRM, RBM, and OPRM systems are the only nuclear instrumentation systems that utilize the LPRM readings. The APRM, RBM and OPRM system responses will not be impacted by the LPRM calibration interval extension.

The APRM system provides indication of core average thermal power and input to the Reactor Protection System (RPS). Above 23% rated thermal power (RTP), the APRM

readings are adjusted to conform to the reactor power calculated from a heat balance. The adjustment is required by SR 3.3.1.1.3 on a weekly basis. APRM signals used by the RPS are adjusted to conform to reactor power more frequently than either the current or proposed LPRM calibration interval; therefore, the APRM signals are not impacted by the calibration interval extension.

The RBM system limits non-peripheral control rod withdrawal if changes in localized neutron flux during control rod manipulations exceed a predefined setpoint, as identified in the Core Operating Limits Report (COLR). The RBM system is assumed to function to block further control rod withdrawal to preclude a Safety Limit Minimum Critical Power Ratio (SLMCPR) violation. The OPRM system is capable of detecting thermal hydraulic instability by monitoring changes in local neutron flux within the reactor core. The OPRM system also provides input to the RPS. The RBM and OPRM systems rely on relative changes in LPRM signals. These systems are not impacted by increased LPRM signal uncertainty, due to LPRM calibration interval extension, because this LPRM signal uncertainty cancels out of the signal ratios utilized by these systems.

BSEP, Units 1 and 2 use the POWERPLEX-III Core Monitoring Software System (CMSS). This system is a group of computer programs that, together with inputs from the nuclear instrumentation, provide continuous online core monitoring. The POWERPLEX-III CMSS continuously monitors LPRM detectors for drift. Baseline LPRM readings used by the drift detection algorithm are reset following changes in operating conditions such as reactor power, core flow, and control rod motion. These changes are independent of the LPRM calibration interval and occur more frequently than either the current or the extended LPRM calibration interval. Therefore, the capability of the core monitoring system to identify drifting LPRM detectors is not impacted by the LPRM calibration interval extension.

LPRM inputs to the POWERPLEX-III CMSS are used by the core monitoring system to calculate the core power distribution and margins to established fuel thermal operating limits that are monitored to verify operation in compliance with TS.

As described above, LPRM gain settings are determined from the local flux profiles measured by the TIP system to calibrate the LPRM detectors. The current SR 3.3.1.1.8 frequency interval of 1100 MWD/T between required LPRM calibrations is based on operating experience with older design LPRM detectors at BSEP, Units 1 and 2 and use of an LPRM signal uncertainty that is consistent with the 1100 MWD/T frequency interval to determine the power distribution uncertainty used in SLMCPR analysis. Extending the LPRM calibration surveillance interval will increase LPRM signal uncertainty.

BSEP currently uses newer design LPRM chambers that exhibit more consistent sensitivity behavior than previous LPRM detectors. The current reload core designs (i.e.,

Unit 1 Cycle 17 and Unit 2 Cycle 19) are licensed using the NRC approved AREVA design methodologies as identified in TS 5.6.5, "Core Operating Limits Report (COLR)." To support an extension of the LPRM calibration interval, BSEP requested that AREVA increase the LPRM signal uncertainty used to determine the power distribution uncertainty applied in the current Unit 1 and Unit 2 SLMCPR analyses. The LPRM signal uncertainty was increased to a value consistent with an extended LPRM calibration interval of 2500 EFPH.

Extending the LPRM calibration surveillance interval will increase the LPRM signal uncertainty. A plant-specific LPRM uncertainty analysis for a 2000 EFPH LPRM calibration frequency has been performed, and the results are documented in Enclosure 6 to this letter. The analysis confirmed that the increased LPRM signal uncertainty used to determine the power distribution uncertainty applied in the current SLMCPR analyses remains bounding for BSEP if the LPRM calibration interval is extended to 2000 EFPH. BSEP fuel reload licensing analyses assume 4.30% uncertainty in the LPRM readings used by the core monitoring system. This uncertainty bounds the BSEP-specific LPRM uncertainty of 3.56% following the proposed 2000 EFPH calibration interval.

An additional analysis was performed to evaluate the plant-specific LPRM signal uncertainty when accounting for the SR 3.0.2 allowed 25% extension of the calibration interval (i.e., resulting in a calibration interval of 2500 EFPH). The results of this analysis, which is also provided in Enclosure 6, demonstrate that LPRM signal uncertainty following the maximum calibration interval of 2500 EFPH is 4.19%, which remains bounded by the LPRM signal uncertainty included in the current SLMCPR analyses.

Thus, an increase in the calibration interval from 1100 MWD/T to 2000 EFPH is bounded by the uncertainties currently applied to the licensing basis analyses. The LPRM detector decay constant and constant sensitivity exposure threshold on which the revised BSEP-specific LPRM signal uncertainties are based are installed in the BSEP Unit 1 and Unit 2 core monitoring systems.

As part of the proposed change, the revised LPRM calibration frequency is expressed in terms of effective full power hours, rather than the existing megawatt days per metric ton core average exposure. The difference between the exposure units of EFPH and MWD/MTU is small. The conversion value for BSEP is approximately 0.824 EFPH per MWD/T, with small cycle-to-cycle deviation based on the amount of uranium in the core.

Conclusion

The AREVA reload licensing analyses of the BSEP Unit 1 and Unit 2 cores consider an LPRM detector signal uncertainty of 4.30%. The analysis in Enclosure 6 demonstrates a BSEP LPRM signal uncertainty of 4.19% for a maximum LPRM calibration interval of

2500 EPFH (i.e., 2000 EFPH plus the 25% extension of the calibration interval allowed by SR 3.0.2). Therefore, the calibration interval extension will not affect any safety analysis methods, core thermal limits documented in the COLR, or the current safety analysis results as documented in the BSEP Unit 1 and 2 Updated Final Safety Analysis Report (UFSAR).

Extending the LPRM calibration surveillance interval will increase the LPRM signal uncertainty. However, this increase in the LPRM signal uncertainty value is justified based on a plant-specific evaluation that confirms that the LPRM signal uncertainty resulting from the extended surveillance interval is bounded by the value used to determine the power distribution uncertainty applied in BSEP SLMCPR analyses.

4.0 Regulatory Evaluation

4.1 Applicable Regulatory Requirements/Criteria

As stated in the NRC's "Safety Evaluation of the Brunswick Steam Electric Station Units 1 and 2," dated November 1973 (i.e., Reference 3), BSEP meets the intent of the General Design Criteria (GDC), published in the Federal Register on May 21, 1971, as Appendix A to 10 CFR Part 50. The proposed changes do not affect compliance with the GDCs. In particular, the intent of GDC 10, "Reactor Design," GDC 13, " Instrumentation and Control," and GDC 20, "Protection System Functions," continue to be met.

10 CFR 50.36(c)(3), "Surveillance requirements," states that SRs are requirements relating to the test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation (LCO) will be met.

The proposed change is a result of increasing the surveillance interval of the LPRM calibration frequency from 1100 MWD/T to 2000 EFPH average core exposure. Increasing the frequency interval between required LPRM calibrations is acceptable due to improvements in the fuel analytical bases. Therefore, the revised surveillance interval continues to ensure that the LPRM detector signal is adequately calibrated. Extending the LPRM calibration surveillance interval will increase the LPRM signal uncertainty value used in the BSEP, Unit 1 and 2 SLMCPR analysis; however, this increase in the LPRM signal uncertainty value is acceptable since the increase is bounded by the values used by the AREVA analysis. This calibration continues to provide assurance that the LPRM accuracy remains within that used to determine the total power uncertainty assumed in the thermal analysis basis; and, therefore, the LCO will continue to be met.

10 CFR 50 Appendix A, General Design Criterion (GDC) 10, "Reactor Design," states that the reactor protection systems shall 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. The proposed change has no impact on equipment design or fundamental operation and LPRM accuracy remains within that used to determine the total power uncertainty assumed in the thermal analysis basis, therefore maintaining thermal limits and the safety margin. As such, the intent of GDC 10 continues to be met.

10 CFR 50, Appendix A, GDC 13, " Instrumentation and Control," states that instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, anticipated operational occurrences, and accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within the prescribed operating ranges. The proposed change does not affect the control parameters governing unit operation and LPRM accuracy remains within that used to determine the total power uncertainty assumed in the thermal analysis basis. Therefore, the instrumentation and controls will continue to meet the intent of GDC 13.

10 CFR 50, Appendix A, GDC 20, "Protection System Functions," states that the protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences, and (2) to sense accident conditions and to initiate the operation of systems and components important to safety. LPRM accuracy remains within that used to determine the total power uncertainty assumed in the thermal analysis basis and the APRM system adjustment interval is not reduced, allowing the APRM system to continue functioning properly. Therefore, the proposed change will continue to meet the intent of GDC 20.

In conclusion, based on the considerations discussed above, (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.

CP&L has determined that the proposed changes do not require any exemptions or relief from regulatory requirements, other than the TS, and do not affect conformance with any GDC differently than described in the Updated Final Safety Analysis Report (UFSAR).

4.2 Precedent

The NRC has approved similar amendments for James A. Fitzpatrick Nuclear Power Plant (i.e., Reference 4), Vermont Yankee Nuclear Power Plant (i.e., Reference 5), Grand

Gulf Nuclear Power Station (i.e., Reference 6), River Bend Station (i.e., Reference 7), and Peach Bottom Atomic Power Station (i.e., Reference 8), and requested by LaSalle County Station (i.e., Reference 9). The Grand Gulf and LaSalle submittals included a plant-specific statistical evaluation which confirmed that the uncertainty associated with the extended LPRM calibration interval is valid, similar to that provided for BSEP, Units 1 and 2.

4.3 No Significant Hazards Consideration

The proposed change will revise Technical Specification (TS) 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Surveillance Requirement (SR) 3.3.1.1.8 to increase the frequency interval between Local Power Range Monitor (LPRM) calibrations from 1100 megawatt-days per metric ton (MWD/T) average core exposure to 2000 effective full power hours (EFPH). Extending the LPRM calibration surveillance interval will increase the LPRM signal uncertainty. However, this increase in the LPRM signal uncertainty is acceptable due to improvements in fuel analytical bases, core monitoring processes, and nuclear instrumentation and because the increase is bounded by the increased LPRM signal uncertainty value used to determine the power distribution uncertainties applied in Safety Limit Minimum Critical Power Ratio (SLMCPR) analyses. The calibration interval extension will not affect any safety analysis methods, core thermal limits documented in the Core Operating Limits Report, or the current safety analysis results as documented in the BSEP Unit 1 and 2 Updated Final Safety Analysis Report (UFSAR).

According to 10 CFR 50.92, "Issuance of amendment," paragraph (c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

CP&L has evaluated whether or not a significant hazards consideration is involved with the proposed amendments by focusing on the three standards set forth in 10 CFR 50.92, as discussed below.

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The proposed amendments revise the surveillance interval for the LPRM calibration from 1100 MWD/T average core exposure to 2000 effective full power hours (EFPH). Increasing the frequency interval between required LPRM calibrations is acceptable due to improvements in fuel analytical bases, core monitoring processes, and nuclear instrumentation. The revised surveillance interval continues to ensure that the LPRM detector signal will continue to be adequately calibrated.

   This change will not alter the operation of process variables, structures, systems, or components as described in the Updated Final Safety Analysis Report. The probability of an evaluated accident is derived from the probabilities of the individual precursors to that accident. The proposed change does not alter the initiation conditions or operational parameters for the LPRM subsystem and there is no new equipment introduced by the extension of the LPRM calibration interval. The performance of the Average Power Range Monitor (APRM), Rod Block Monitor (RBM), and Oscillation Power Range Monitor (OPRM) systems is not affected by the proposed surveillance interval increase. The proposed LPRM calibration interval extension will have no significant effect on the Reactor Protection System (RPS) instrumentation accuracy during power maneuvers or transients and will, therefore, not significantly affect the performance of the RPS. As such, no individual precursors of an accident are affected and the proposed amendments do not increase the probability of a previously analyzed event.

   The radiological consequences of an accident can be affected by the thermal limits existing at the time of the postulated accident; however, increasing the surveillance interval frequency will not increase the calculated thermal limits since all uncertainties associated with the increased interval are currently implemented and are currently used to calculate the existing safety limits. Plant specific evaluation of LPRM sensitivity to exposure has determined that the extended calibration frequency increases the LPRM signal uncertainty value used in the SLMCPR analysis; however, the increase is bounded by the values currently used in the safety analysis. Therefore, the thermal limit calculation is not significantly affected by LPRM calibration frequency, and thus the radiological consequences of any accident previously evaluated are not increased.

   Based on the above, the proposed amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   Creation of the possibility of a new or different kind of accident requires creating one or more new accident precursors. New accident precursors may be created by modifications of plant configuration, including changes in allowable modes of operation. The performance of the APRM, RBM, and OPRM systems are not affected by the proposed LPRM surveillance interval increase. The proposed change does not affect the control parameters governing unit operation or the response of plant equipment to transient conditions. For the proposed LPRM extended calibration interval frequency, all uncertainties remain less than the uncertainties assumed in the existing thermal limit calculations. The proposed change does not change or introduce any new equipment, modes of system operation, or failure mechanisms; therefore, no new accident precursors are created. Based on the above information, the proposed amendments do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No

   The proposed change has no impact on equipment design or fundamental operation, and there are no changes being made to safety limits or safety system allowable values that would adversely affect plant safety as a result of the proposed LPRM surveillance interval increase. The performance of the APRM, RBM, and OPRM systems are not affected by the proposed change. The margin of safety can be affected by the thermal limits existing at the time of the postulated accident; however, uncertainties associated with LPRM chamber exposure have no significant effect on the calculated thermal limits. Plant-specific evaluation of LPRM sensitivity to exposure has determined that the extended calibration frequency increases the LPRM signal uncertainty value used in the SLMCPR analysis; however, the increase is bounded by the values currently used in the safety analysis. The thermal limit calculation is not significantly affected since LPRM sensitivity with exposure is well defined. LPRM accuracy remains within that used to determine the total power uncertainty assumed in the thermal analysis basis, therefore maintaining thermal limits and the safety margin. The proposed change does not affect uncertainties or initial conditions assumed in the thermal limit calculations and therefore the margin of safety in the safety analyses is maintained. Based on the above information, the proposed amendments do not result in a significant reduction in the margin of safety.

Based on the above, CP&L concludes that the proposed amendments present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

6.0 Environmental Considerations

A review has determined that the proposed amendments would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, "Standards for Protection Against Radiation," or would change an inspection or surveillance requirement. However, the proposed amendments do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendments meet the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9), "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review." Therefore, in accordance with 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendments.

7.0 References

1. Letter from Farideh E. Saba (USNRC) to Benjamin Waldrep (CP&L), *Issuance of Amendments to Support Transition to AREVA Fuel and Methodologies (TAC Nos. MD4063 and MD4064)*, March 7, 2008, ADAMS Accession Number ML080870478.

2. Letter from James Scarola (CP&L) to U.S. Nuclear Regulatory Commission, *Request for License Amendments Regarding Linear Heat Generation Rate and Core Operating Limits Report References for AREVA NP Fuel*, January 22, 2007, ADAMS Accession Number ML070300570.

3. Letter from NRC to Michael Kansler (Entergy Nuclear Operations, Inc.), *James A. Fitzpatrick Nuclear Power Plant – Amendment Re: Regarding Local Power Range Monitor Calibration Frequency (TAC No. MB6945)*, dated May 1, 2003, ADAMS Accession Number ML030860088.

4. NRC "Safety Evaluation of the Brunswick Steam Electric Station Units 1 and 2," dated November 1973.

5. Letter from NRC to Samuel L. Newton (Vermont Yankee Nuclear Power Corporation), *Vermont Yankee Nuclear Power Station – Issuance of Amendment*

Re: Local Power Range Monitor Calibration Frequency (TAC No. MA9053), dated July 18, 2000, ADAMS Accession Number ML003733066.

6. Letter from NRC to William R. Brian (Entergy Operations, Inc.), *Grand Gulf Nuclear Station, Unit 1 – Issuance of Amendment Re: Changes to Technical Specifications Surveillance Requirement 3.3.1.1.7, The Local Power Range Monitor Calibration Frequency (TAC No. MD3469)*, dated October 24, 2007, ADAMS Accession Number ML073190250.

7. Letter from NRC to Randall K. Edington (Entergy Operations, inc.), *River Bend Station, Unit 1 – Issuance of Amendment Re: Changes to Local Power Range Monitor (LPRM) Calibration Frequency (TAC No. M98883)*, dated June 11, 1999, ADAMS Accession Number ML021620290.

8. Letter from NRC to Charles G. Pardee (Exelon Generation Company, LLC), *Peach Bottom Atomic Power Station, Units 2 and 3 – Issuance of Amendments to Extend Local Power Range Monitor Calibration Interval (TAC Nos. MD3717 and MD3718)*, dated February 29, 2008, ADAMS Accession Number ML080390032.

9. Letter from Patrick R. Simpson (Exelon Generation Company, LLC) to NRC, *Request for a License Amendment to Revise Local Power Range Monitor Calibration Frequency*, dated July 25, 2008, ADAMS Accession Number ML082110187.

Marked-up Technical Specification Pages - Unit 1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.5	Perform a functional test of each automatic scram contactor.	7 days
SR 3.3.1.1.6	Verify the source range monitor (SRM) and intermediate range monitor (IRM) channels overlap.	Prior to withdrawing SRMs from the fully inserted position
SR 3.3.1.1.7	----------NOTE---------- Only required to be met during entry into MODE 2 from MODE 1. ---------- Verify the IRM and APRM channels overlap.	7 days
SR 3.3.1.1.8	Calibrate the local power range monitors.	~~1100 MWD/T average core exposure~~
SR 3.3.1.1.9	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.1.1.10	Calibrate the trip units.	92 days

(continued)

2000 effective full power hours

Typed Technical Specification Pages - Unit 1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.5	Perform a functional test of each automatic scram contactor.	7 days
SR 3.3.1.1.6	Verify the source range monitor (SRM) and intermediate range monitor (IRM) channels overlap.	Prior to withdrawing SRMs from the fully inserted position
SR 3.3.1.1.7	----------NOTE---------- Only required to be met during entry into MODE 2 from MODE 1. ---------- Verify the IRM and APRM channels overlap.	7 days
SR 3.3.1.1.8	Calibrate the local power range monitors.	2000 effective full power hours
SR 3.3.1.1.9	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.1.1.10	Calibrate the trip units.	92 days

(continued)

Typed Technical Specification Page - Unit 2

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.5	Perform a functional test of each automatic scram contactor.	7 days
SR 3.3.1.1.6	Verify the source range monitor (SRM) and intermediate range monitor (IRM) channels overlap.	Prior to withdrawing SRMs from the fully inserted position
SR 3.3.1.1.7	----------NOTE---------- Only required to be met during entry into MODE 2 from MODE 1. ---------- Verify the IRM and APRM channels overlap.	7 days
SR 3.3.1.1.8	Calibrate the local power range monitors.	2000 effective full power hours
SR 3.3.1.1.9	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.1.1.10	Calibrate the trip units.	92 days

(continued)

Marked-up Technical Specification Bases Page - Unit 1
(For information only)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.6 and SR 3.3.1.1.7 (continued)

the transition between MODE 1 and MODE 2 can be made without either APRM downscale rod block, or IRM upscale rod block. Overlap between SRMs and IRMs similarly exists when, prior to withdrawing the SRMs from the fully inserted position, IRM readings have doubled before the SRMs have reached the high-high upscale trip.

As noted, SR 3.3.1.1.7 is only required to be met during entry into MODE 2 from MODE 1. That is, after the overlap requirement has been met and indication has transitioned to the IRMs, maintaining overlap is not required (APRMs may be reading downscale once in MODE 2).

If overlap for a group of channels is not demonstrated (e.g., IRM/APRM overlap), the reason for the failure of the Surveillance should be determined and the appropriate channel(s) declared inoperable. Only those appropriate channels that are required in the current MODE or condition should be declared inoperable.

A Frequency of 7 days is reasonable based on engineering judgment and the reliability of the IRMs and APRMs.

SR 3.3.1.1.8

LPRM gain settings are determined from the local flux profiles measured by the Traversing Incore Probe (TIP) System. This establishes the relative local flux profile for appropriate representative input to the APRM System. ~~The 1100 MWD/T Frequency is based on operating experience with LPRM sensitivity changes. The core weight, tons (T) in MWD/T, reflects metric tons.~~

SR 3.3.1.1.9 and SR 3.3.1.1.12

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the channel will perform the intended function. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The 92 day Frequency of SR 3.3.1.1.9 is based on the reliability analysis of Reference 11.

(continued)

The calibration interval of 2000 effective full power hours (EFPH) is related to the current licensed core thermal power in that it should be adjusted in inverse proportion to increases in licensed rated core thermal power so that core average detector fluence accumulated during the calibration interval remains essentially unchanged. The 2000 EFPH Frequency is based on LPRM operating experience and that the LPRM detector uncertainty remains less than that used to determine power distribution uncertainty applied in the calculation of the Safety Limit Minimum Critical Power Ratio (SLMCPR), as shown in Reference 23.

BASES

2000 EFPH

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.13 (continued)

calorimetric calibration (SR 3.3.1.1.3) and the ~~1100 MWD/T~~ LPRM calibration against the TIPs (SR 3.3.1.1.8).

A second Note is provided that requires the IRM SRs to be performed within 12 hours of entering MODE 2 from MODE 1. Testing of the MODE 2 IRM Functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or movable links. This Note allows entry into MODE 2 from MODE 1 if the associated Frequency is not met per SR 3.0.2. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

A third note is provided that requires that the recirculation flow (drive flow) transmitters, which supply the flow signal to the APRMs, be included in the SR for Functions 2.b and 2.f. The APRM Simulated Thermal Power—High Function (Function 2.b) and the OPRM Upscale Function (Function 2.f) both require a valid drive flow signal. The APRM Simulated Thermal Power—High Function uses drive flow to automatically enable or bypass the OPRM Upscale trip output to RPS. A CHANNEL CALIBRATION of the APRM drive flow signal requires both calibrating the drive flow transmitters and the processing hardware in the APRM equipment. SR 3.3.1.1.18 establishes a valid drive flow/core flow relationship. Changes throughout the cycle in the drive flow/core flow relationship due to the changing thermal hydraulic operating conditions of the core are accounted for in the margins included in the bases or analyses used to establish the setpoints for the APRM Simulated Thermal Power—High Function and the OPRM Upscale Function.

The Frequency of SR 3.3.1.1.13 is based upon the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.1.1.14

(Not used.)

(continued)

BASES

REFERENCES (continued)

22. General Electric Nuclear Energy Letter NSA 01-212, DRF C51-00251-00, A. Chung (GE) to S. Chakraborty (GE), "Minimum Number of Operable OPRM Cells for Option III Stability at Brunswick 1 and 2," June 8, 2001.

23. Calculation 0B21-1305, Core Monitoring LPRM Uncertainty and Sensitivity Decay.