February 9, 2007

Mr. J. V. Parrish
Chief Executive Officer
Energy Northwest
P.O. Box 968 (Mail Drop 1023)
Richland, WA 99352-0968

SUBJECT: COLUMBIA GENERATING STATION - ISSUANCE OF AMENDMENT RE: SUPPRESSION CHAMBER-TO-DRYWELL VACUUM BREAKERS AND DRYWELL-TO-SUPPRESSION CHAMBER BYPASS LEAKAGE TEST (TAC NO. MD1225)

Dear Mr. Parrish:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 201 to Facility Operating License No. NPF-21 for the Columbia Generating Station. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated April 18, 2006.

The amendment revises TS Surveillance Requirement (SR) 3.6.1.1.2 by changing the test frequency of the drywell-to-suppression chamber bypass leakage test from 24 months to 120 months. The amendment also adds new TS SRs 3.6.1.1.3 and 3.6.1.1.4, to test the suppression chamber-to-drywell vacuum breakers on a 24-month frequency.

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

Sincerely,

/RA/

Carl F. Lyon, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-397

Enclosures:
1. Amendment No. 201 to NPF-21
2. Safety Evaluation

cc w/encls: See next page

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cc:
Mr. W. Scott Oxenford (Mail Drop PE04)
Vice President, Technical Services
Energy Northwest
P.O. Box 968
Richland, WA 99352-0968

Mr. Albert E. Mouncer (Mail Drop PE01)
Vice President, Corporate Services/
General Counsel/CFO
Energy Northwest
P.O. Box 968
Richland, WA 99352-0968

Chairman
Energy Facility Site Evaluation Council
P.O. Box 43172
Olympia, WA 98504-3172

Mr. Douglas W. Coleman (Mail Drop PE20)
Manager, Regulatory Programs
Energy Northwest
P.O. Box 968
Richland, WA 99352-0968

Mr. Gregory V. Cullen (Mail Drop PE20)
Supervisor, Licensing
Energy Northwest
P.O. Box 968
Richland, WA 99352-0968

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011-4005

Chairman
Benton County Board of Commissioners
P.O. Box 190
Prosser, WA 99350-0190

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
P.O. Box 69
Richland, WA 99352-0069

Mr. Dale K. Atkinson (Mail Drop PE08)
Vice President, Nuclear Generation
Energy Northwest
P.O. Box 968
Richland, WA 99352-0968

Mr. William A. Horin, Esq.
Winston & Strawn
1700 K Street, N.W.
Washington, DC 20006-3817

Mr. Matt Steuerwalt
Executive Policy Division
Office of the Governor
P.O. Box 43113
Olympia, WA 98504-3113

Ms. Lynn Albin
Washington State Department of Health
P.O. Box 7827
Olympia, WA 98504-7827

Technical Services Branch Chief
FEMA Region X
130 228th Street, S.W.
Bothell, WA 98201-9796

Ms. Cheryl M. Whitcomb (Mail Drop PE03)
Vice President, Organizational
Performance & Staffing/CKO
Energy Northwest
P.O. Box 968
Richland, WA 99352-0968

Assistant Director
Nuclear Safety and Energy Siting Division
Oregon Department of Energy
625 Marion Street, NE
Salem, OR 97301-3742

ENERGY NORTHWEST

DOCKET NO. 50-397

COLUMBIA GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 201
License No. NPF-21

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

   A. The application for amendment by Energy Northwest (licensee), dated April 18, 2006, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications and paragraph 2.C.(2) of Facility Operating License No. NPF-21 as indicated in the attachment to this license amendment.

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

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

David Terao, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Facility
Operating License and
Technical Specifications

Date of Issuance: February 9, 2007

ATTACHMENT TO LICENSE AMENDMENT NO. 201

FACILITY OPERATING LICENSE NO. NPF-21

DOCKET NO. 50-397

Replace the following pages of the Facility Operating License No. NPF-21 and Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Facility Operating License

REMOVE	INSERT
- 3 -	- 3 -

Technical Specifications

REMOVE	INSERT
3.6.1.1-2	3.6.1.1-2
------	3.6.1.1-3

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 201 TO

FACILITY OPERATING LICENSE NO. NPF-21

ENERGY NORTHWEST

COLUMBIA GENERATING STATION

DOCKET NO. 50-397

1.0 INTRODUCTION

By application dated April 18, 2006 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML061110163), Energy Northwest (the licensee) requested changes to the Technical Specifications (TSs) for the Columbia Generating Station (Columbia). The requested TS changes would revise requirements concerning drywell-to-suppression chamber bypass leakage. Specifically, the licensee has proposed to: (1) extend the baseline interval specified in Surveillance Requirement (SR) 3.6.1.1.2 for the drywell-to-suppression chamber bypass leakage test from 24 months to 120 months, and (2) add new SR 3.6.1.1.3 and SR 3.6.1.1.4, which would establish specific testing requirements for the suppression chamber-to-drywell vacuum breakers.

Drywell-to-suppression chamber bypass leakage pathways can be grouped into three categories: (1) the drywell floor and drywell floor penetrations, (2) piping externally connected to both the drywell and suppression chamber airspace, and (3) the suppression chamber-to-drywell vacuum breakers. The licensee's proposed TS changes would effectively extend the bypass leakage testing interval for the first two categories, while maintaining the current surveillance interval for the vacuum breakers.

2.0 REGULATORY EVALUATION

2.1 Background

Columbia is a General Electric boiling-water reactor, class 5 (BWR/5) plant with Mark II primary containment. The Mark II containment consists of two compartments, the drywell and the suppression chamber. The drywell has the shape of a truncated cone, and is located above the cylindrically shaped suppression chamber. The drywell floor separates the drywell and suppression chamber, forming the primary boundary to bypass leakage.

During a design-basis loss-of-coolant accident, steam would enter the drywell, resulting in a relatively rapid increase in drywell-to-suppression chamber differential pressure. Once this differential pressure exceeds approximately 6.4 pounds per square inch differential (psid), the steam and gases in the drywell atmosphere would relieve through the downcomers which penetrate the drywell floor, and be discharged into the suppression chamber pool. The cool water in the suppression chamber pool would then condense the steam to limit the pressure inside primary containment to less than its design value. For the pressure-suppression feature of the Mark II containment to be effective, an excessive amount of steam must not bypass the intended condensation pathway by leaking directly to the suppression chamber airspace. Thus, the basis for bypass leakage testing requirements is to verify the functionality of the pressure-suppression feature of containment, in order to assure containment integrity.

Maintaining the pressure suppression function of primary containment requires limiting the leakage from the drywell to the suppression chamber. TS SR 3.6.1.1.2 measures drywell-to-suppression chamber differential pressure to ensure that the leakage paths that would bypass the suppression pool are within allowable limits. The containment functional design is further described in Final Safety Analysis Report (FSAR) Section 6.2.1, and containment leakage testing is further described in FSAR Section 6.2.6.

The drywell-to-suppression chamber bypass leakage test is conducted as an individual test or as part of the Primary Containment Leakage Rate Testing Program for Type A Tests (i.e., Integrated Leakage Rate Test). The frequency of Type A Tests at Columbia is in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix J, Option B, "Performance-Based Requirements."

2.2 Proposed TS Changes

The licensee's proposed TS changes would extend the bypass leakage testing interval for the drywell floor, drywell floor penetrations, and piping lines externally connecting the drywell to the suppression chamber airspace, while maintaining the current interval for the suppression chamber-to-drywell vacuum breakers. However, the acceptance criteria in proposed SR 3.6.1.1.3 and SR 3.6.1.1.4 are new; currently, bypass leakage testing for all pathways is performed simultaneously, and thus no specific acceptance criteria are established for the vacuum breakers. The proposed changes are similar to TS changes approved by the Nuclear Regulatory Commission (NRC) for LaSalle County Station, Units 1 and 2, on November 7, 2001 (ADAMS Accession No. ML012850399).

2.2.1 Proposed Extension to Test Interval of SR 3.6.1.1.2

SR 3.6.1.1.2 reads:

> Verify drywell to suppression chamber bypass leakage rate is less than or equal to the equivalent leakage rate through an orifice 0.005 ft^2 at an initial differential pressure of $\geq$ 1.5 psid.

Currently, the surveillance frequency for SR 3.6.1.1.2 is 24 months and, after two consecutive test failures, SR 3.6.1.1.2 must be performed each 12 months, until two consecutive tests pass.

The licensee proposes to revise SR 3.6.1.1.2 to read:

> Verify drywell to suppression chamber bypass leakage is ≤ 10% of the acceptable A/√k design value of 0.050 ft^2 at an initial differential pressure of ≥ 1.5 psid.

The licensee has proposed to change the frequency for SR 3.6.1.1.2 to the following:

> 120 months,
>
> <u>AND</u>
>
> 48 months following a test with bypass leakage greater than the bypass leakage limit,
>
> <u>AND</u>
>
> 24 months following two consecutive tests with bypass leakage greater than the bypass leakage limit until two consecutive tests are less than or equal to the bypass leakage limit.

2.2.2 <u>Proposed Addition of SR 3.6.1.1.3 and SR 3.6.1.1.4</u>

The licensee has proposed to add SR 3.6.1.1.3, which would read:

> Verify individual drywell to suppression chamber vacuum relief valve bypass pathway leakage is ≤ 1.2% of the acceptable A/√k design value of 0.050 ft^2 at an initial differential pressure of ≥ 1.5 psid.

The licensee has proposed to add SR 3.6.1.1.4, which would read:

> Verify total drywell to suppression chamber vacuum relief valve bypass leakage is ≤ 3.0% of the acceptable A/√k design value of 0.050 ft^2 at an initial differential pressure of ≥ 1.5 psid.

Both proposed SRs would be performed with a 24-month interval. A note would precede each SR, indicating that the performance of SR 3.6.1.1.2 satisfies SR 3.6.1.1.3 and SR 3.6.1.1.4. The licensee states that the note is included since drywell-to-suppression chamber vacuum relief valve leakage is included in the measurement of the drywell-to-suppression chamber bypass leakage required by SR 3.6.1.1.2.

3.0 <u>TECHNICAL EVALUATION</u>

This evaluation consists of two parts: (1) an analysis of the proposed surveillance interval extension for SR 3.6.1.1.2, and (2) an analysis of the proposed acceptance criteria for SR 3.6.1.1.3 and SR 3.6.1.1.4.

3.1 Analysis of the Proposed Surveillance Interval Extension for SR 3.6.1.1.2

3.1.1 Leakage Pathways Affected by Proposed Surveillance Interval Extension for SR 3.6.1.1.2

Two categories of leakage pathways would be affected by the proposed surveillance interval extension for SR 3.6.1.1.2. These are: (1) the drywell floor and its penetrations, and (2) piping which is externally cross-connected between the drywell and suppression chamber airspace.

3.1.1.1 Drywell Floor and Floor Penetrations

The drywell floor, which separates the drywell and suppression chamber, is a 2-foot thick slab of reinforced concrete, which is supported by structural steel beams in composite action, by reinforced-concrete columns, and by a 5-foot inner circular reinforced-concrete slab, inside the reactor pedestal. The drywell floor structural integrity proof test was performed after initial completion of the initial Integrated Leak Rate Test (ILRT) to verify acceptable quality, structural integrity, and leak-tightness. Drywell floor penetrations include the downcomers, safety-relief valve (SRV) discharge lines, and instrument penetrations for use during outages.

There are 102 carbon steel downcomers connecting the drywell to the suppression chamber pool, of which 84 are 24-inch outside diameter (OD) and 18 are 28-inch OD. Three of these downcomers (one 24-inch and two 28-inch) have been capped. Eighteen 10-inch, carbon steel SRV discharge lines penetrate the 28-inch OD downcomer jet deflector plates (which include the two capped 28-inch downcomers) and then penetrate the downcomer pipe wall below the drywell floor and terminate in the suppression chamber pool of water.

Both the downcomers and the SRV discharge lines were designed to the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section III, “Rules for Construction of Nuclear Facility Components.” Though not required by the ASME Code, the licensee performed a fatigue analysis for the downcomers and SRV discharge lines which confirmed that these lines would maintain their structural integrity under all postulated loading conditions.

The licensee stated in its application that a comprehensive periodic visual examination program for the primary containment structure is already in place and is being implemented as part of the Columbia Inservice Inspection (ISI) Program. Columbia’s ISI Program is intended to identify defects which could jeopardize the leak-tightness and structural integrity of the containment, and it complies with the ASME Code, Section XI. The ISI inspections are required to be conducted three times within each 10-year testing interval.

Based upon the high quality of the components and structures which constitute the drywell floor and penetrations, the licensee’s other inspection and leakage testing requirements and procedures, and the good testing history (see Section 3.1.2, below), the staff finds that a drywell-to-suppression chamber bypass leakage surveillance interval of 120 months is justified for these potential leakage pathways.

3.1.1.2 Externally Cross-Connected Piping

Externally cross-connected piping consists of those lines which connect the drywell and suppression chamber airspace without penetrating the drywell floor. The licensee has identified four systems containing piping that meet this definition:

1. Containment vent and purge lines, including the nitrogen inerting/de-inerting/make-up lines (two flow paths of 24-inch and 30-inch diameter piping, and one flow path of 1-inch diameter piping).

2. Drywell and suppression chamber Residual Heat Removal System spray lines (two flow paths of 16-inch and 6-inch diameter piping).

3. Hydrogen and oxygen analyzer lines (two flow paths of 1.5-inch diameter piping).

4. Hydrogen recombiner lines (two flow paths of 4-inch diameter piping).

The licensee has stated that all cross-connected piping lines have multiple, in-series containment isolation valves which are designed to meet the requirements of Appendix J to 10 CFR Part 50. Periodic local leakage rate testing is performed on these isolation valves, in accordance with the requirements of Appendix J. Leakage rate testing at Columbia is controlled by the Columbia Primary Containment Leak Rate Testing Program Plan.

Due to a design which includes multiple isolation valves in series, and the licensee's leakage rate testing program, the staff finds that the proposed extension of the bypass leakage surveillance interval to 120 months is justified for externally cross-connected piping.

3.1.2 Historical Drywell-to-Suppression Chamber Bypass Leakage Test Results

The drywell-to-suppression chamber bypass leakage test involves the pressurization of the drywell, and it may be performed either individually or concurrently with the ILRT for primary containments required by Appendix J to 10 CFR Part 50. As indicated in its April 18, 2006, submittal, Columbia has performed a total of 23 drywell-to-suppression chamber bypass leakage tests, in which all of the tests had successful results with significant margin. The initial tests were performed in 1984 during preoperational testing at differential pressures of 25, 15, 5, and 1.5 psid. Subsequently, Columbia has performed three additional tests at 5 psid and 17 tests at 1.5 psid.

As expected, the highest leakage was recorded during the test conducted at the highest differential pressure of 25 psid (in 1984). That test resulted in leakage of 28.3 weight percent per day (wt%/day), which is 22 percent of the test acceptance criteria of 128.56 weight percent per day. The test acceptance criterion of 128.56 weight percent per day is based on an equivalent orifice size of 0.0045 square feet (sq. ft). The licensee uses conservatism with the test acceptance criterion since it is only 9 percent of the 0.05 sq. ft. design-basis value.

Table 1, below, is the NRC staff's summary of the bypass leakage test results provided by the licensee for Columbia. Table 1 demonstrates that bypass leakage has consistently been a relatively small percentage of the TS allowable leakage, and that design limits have not been approached. Therefore, the NRC staff finds that past test results support the proposed extension to the bypass leakage surveillance interval for the drywell floor, drywell floor penetrations, and externally cross-connected piping.

Table 1: Summary of Test Results for Bypass Leakage Test

AVERAGE MEASURED BYPASS LEAKAGE VALUES (1.5 psid)	
Number of Tests	17
% of Test Acceptance Criteria (78.4 wt%/day)	12.1%
% of Design Basis Value (871.1 wt%/day)	1.1%

3.1.3 Containment Over-Pressurization Analysis

As described in Section 2.0 above, performance of SR 3.6.1.1.2 is intended to verify that drywell-to-suppression chamber bypass leakage pathways are not large enough such that the bypass leakage due to an analyzed accident could result in the over-pressurization of the primary containment. Therefore, it is necessary to consider the potential effect upon the primary containment over-pressurization frequency due to the proposed increase to the surveillance interval for SR 3.6.1.1.2.

The licensee stated in its application that the dominant contributor to the failure of the primary containment pressure-suppression function is the suppression chamber-to-drywell vacuum breakers failing to operate as designed. The drywell floor, its penetrations, and cross-connected piping are reliable passive features which make relatively small contributions to the containment over-pressurization frequency. As explained in Section 2.2 of this evaluation, the proposed surveillance interval extension would only apply to these passive barriers to bypass leakage (i.e., all barriers except the vacuum breakers). Therefore, the licensee's proposal would not be expected to significantly increase the frequency of large bypass leakage events.

If a significant bypass leakage event were to occur, however, alternate means are available to limit the pressure inside the primary containment. Columbia is equipped with drywell and suppression chamber sprays, which would suppress pressure by condensing steam in the primary containment atmosphere. Procedures direct operators to initiate the suppression chamber sprays when primary containment pressure is between 1.68 pounds per square inch gauge (psig) and 12 psig. The drywell sprays are capable of terminating any analyzed pressure rise in the primary containment. In the unlikely event that the drywell and suppression chamber sprays were to fail to terminate a primary containment pressure rise, procedures would direct operators to emergency depressurize the reactor vessel pressure using the drywell and suppression chamber vents prior to reaching the primary containment pressure limit.

Based upon the above discussion, the NRC staff concludes that the proposed surveillance interval extension to SR 3.6.1.1.2 would not be expected to significantly increase the frequency of large bypass leakage events, and that alternate methods of pressure suppression in the primary containment are available to mitigate a potential containment over-pressurization.

3.2 Analysis of the Proposed Acceptance Criteria for SR 3.6.1.1.3 and SR 3.6.1.1.4

Through the proposed addition of SR 3.6.1.1.3 and SR 3.6.1.1.4, the licensee would maintain the current surveillance frequency of 24 months for suppression chamber-to-drywell vacuum breaker leakage. However, as explained in Section 2.2 of this evaluation, SR 3.6.1.1.3 and SR 3.6.1.1.4 would establish new acceptance criteria for vacuum breaker testing.

3.2.1 Description of Suppression Chamber-to-Drywell Vacuum Breakers

Suppression chamber-to-drywell vacuum breakers are provided to prevent exceeding the drywell floor negative design pressure and backflooding of the suppression chamber pool water into the drywell. At Columbia, there are nine suppression chamber-to-drywell vacuum breakers which are located in the suppression chamber airspace of the primary containment. Each suppression chamber to drywell vacuum relief valve assembly consists of two discs and seats which operate independently.

3.2.2 Analysis of Proposed Acceptance Criteria

The Columbia FSAR does not prescribe a specific design limit for bypass leakage through the suppression chamber-to-drywell vacuum breakers, but rather, a single limit through all leakage pathways. Therefore, by establishing a specific leakage test acceptance criterion for the vacuum breakers, the implicit assumption is that the leakage through all other pathways would be less than the difference between the total bypass leakage limit and the specific leakage limit for the vacuum breakers. Accounting for the factor of ten margin between the design limit and the TS allowable limit, the licensee's proposed SR 3.6.1.1.4 effectively assumes, therefore, that non-vacuum breaker pathways would account for not more than 30 percent of the bypass leakage limit. Based upon the accompanying discussion in Section 3.1 of this evaluation, the staff finds that the proposed SR 3.6.1.1.4 is sufficiently conservative.

Proposed SR 3.6.1.1.3 is intended to ensure that a single vacuum breaker does not account for an excessive proportion of the total TS allowable leakage for all nine sets of vacuum breakers. Based on information provided in Section 2.2.2 of this evaluation, SR 3.6.1.1.3 specifies a leakage limit for each suppression chamber-to-drywell vacuum breaker pathway of less than or equal to 1.2 percent of the bypass leakage limit. The acceptance criterion of proposed SR 3.6.1.1.3 is reasonable and consistent with good maintenance and testing practices; therefore, the NRC staff finds it to be acceptable.

Other BWRs with Mark II containments have similarly designed suppression chamber-to-drywell vacuum breakers installed in their downcomers, have also extended their surveillance interval for the Bypass Leakage Test, and perform individual leakage tests on their valves (e.g., Susquehanna Steam Electric Station, Units 1 and 2 (ADAMS Accession No. ML010110174); Limerick Generating Station, Units 1 and 2 (ADAMS Accession No. ML011560214); and Nine Mile Point Nuclear Station, Unit 2 (ADAMS Accession No. ML011140033)). These test results show that measured leakage has been a small percentage of the allowable leakage. The NRC staff finds the proposed changes to establish leakage limits for the suppression chamber-to-drywell vacuum breakers are acceptable as demonstrated by the results from other Mark II BWR suppression chamber-to-drywell vacuum breaker leakage tests.

3.3 Conclusion

The NRC staff has concluded, based on the analysis in Section 3.0 of this evaluation and the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission’s regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public. Therefore, the staff finds the licensee’s proposed TS changes to be acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes a surveillance requirement. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (71 FR 29674; published on May 23, 2006). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and 10 CFR 51.22(c)(10). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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

Principal Contributor: B. Lee

Date: February 9, 2007