October 26, 2001

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 TO REVISE THE FINAL SAFETY ANALYSIS REPORT TO REFLECT ACTUAL PLANT DESIGN OF THE SPENT FUEL STORAGE AND CASK HANDLING OPERATIONS (TAC NO. MB0573)

Dear Mr. Parrish:

The Commission has issued the enclosed Amendment No. 174 to Facility Operating License No. NPF-21 for the Columbia Generating Station. The amendment consists of changes to the Final Safety Analysis Report (FSAR) in response to your application dated October 30, 2000, as supplemented by letter dated September 13, 2001.

The amendment modified the FSAR to reflect analysis of a HI-STORM 100 spent fuel cask system, spent fuel pool description and crane operations.

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/

Jack Cushing, Project Manager, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-397

Enclosures: 1. Amendment No. 174 to NPF-21
2. Safety Evaluation

cc w/encls: See next page

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ENERGY NORTHWEST

DOCKET NO. 50-397

COLUMBIA GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 174
License No. NPF-21

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

   A. The application for amendment by the Energy Northwest (licensee) dated October 30, 2000, as supplemented by letter dated September 13, 2001, 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, by Amendment No.   , the license is amended to authorize revision of the Final Safety Analysis Report (FSAR), as set forth in the application for amendment by Energy Northwest dated October 30, 2000, and supplement dated September 13, 2001. Energy Northwest shall update the FSAR in accordance with 10 CFR 50.71(e).

3. The license amendment is effective as of its date of issuance and shall be implemented in the next periodic update to the FSAR in accordance with 10 CFR 50.71(e). Implementation of the amendment is the incorporation into the FSAR the changes to the description of the facility as described in the licensee's application dated October 30, 2000, and supplement dated September 13, 2001, and evaluated in the staff's Safety Evaluation attached to this amendment.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by J. Donoghue for/

Stephen Dembek, Chief, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Date of Issuance: October 26, 2001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 174 TO FACILITY OPERATING LICENSE NO. NPF-21

ENERGY NORTHWEST

COLUMBIA GENERATING STATION

DOCKET NO. 50-397

1.0 INTRODUCTION

By application dated October 30, 2000, as supplemented by letter dated September 13, 2001, Energy Northwest (the licensee) requested changes to the Columbia Generating Station Final Safety Analysis Report (FSAR). The proposed changes would revise the FSAR to reflect actual plant design of the spent fuel (SF) storage and cask handling operations. The licensee determined in accordance with 10 CFR 50.59 that an unreviewed safety question (USQ) existed and therefore submitted this license amendment request. The specific changes to the FSAR include:

1. Section 9.1.4.1 of the FSAR, "Fuel Handling System - Design Bases," describes two separate pools for spent fuel handling, when there is only one pool. The FSAR states that there is a spent fuel cask storage and a cask loading pool adjacent to the spent fuel pool. There is not a separate spent fuel cask storage and loading pool. There is a spent fuel cask loading pit located within the spent fuel pool. The proposed change is to eliminate references to separate pools and to add a statement that, "Sufficient redundancy is provided in the reactor building crane such that no credible postulated failure of any crane component will result in dropping of the fuel cask and rupturing the fuel storage pool."

2. The FSAR states that limitations on reactor building crane travel preclude transporting the spent fuel casks over the spent fuel pool. There are no interlocks that prevent crane movement over the spent fuel cask pit loading area, which is part of the spent fuel pool. There are interlocks that prevent movement over the spent fuel racks. The proposed change is to add the statement to the FSAR that, "Interlocks on the reactor building crane prevent travel over the spent fuel racks."

3. The FSAR states that at no time while being transported does the fuel cask pass over any safety related equipment. The cask does pass over a safety-related conduit associated with a fuel pool cooling level instrumentation. The proposed change is to add the statement to the FSAR that, "At no time while being transported does the cask pass over any safe shutdown equipment."

4. The FSAR discusses cask loading, handling, and features of construction associated with the GE IF-300 spent fuel cask rather than the Holtec HI-STORM 100 spent fuel cask system, which is the cask system that will be used. The proposed change would accurately describe the Holtec HI-STORM 100 spent fuel cask system.

The supplemental letter dated September 13, 2001, provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination published in the *Federal Register* on March 21, 2001 (66 FR 15918).

2.0 BACKGROUND

Section 9.1.2 of NUREG-0800, "Standard Review Plan," provides criteria for the staff to review the design and performance of the SF storage and fuel handling systems, respectively. Accordingly, the staff reviews whether (1) the safety function of the spent fuel pool (SFP) and storage racks is maintained, (2) the SF assemblies are in a safe and subcritical array during all credible storage conditions, and (3) a safe means of loading the assemblies into shipping casks is provided.

Section 9.1.4, "Light Load Handling System," of NUREG-0800 also provides criteria for the staff to review the design and performance of the light load handling system used to move the spent fuel assembly into the shipping cask. Staff application of the criteria is focused on, among other things, the adequacy of the methods and equipment for transferring stored fuel to the SF shipping cask. Section 9.1.5, "Overhead Heavy Load Handling Systems," of NUREG-0800 also provides criteria for the staff to review the design and performance of the overhead heavy load handling systems used in moving all heavy loads, i.e., loads weighing more than one fuel assembly and its associated handling device. Staff application of the criteria in Standard Review Plan (SRP) Section 9.1.5 is in accordance with NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," July 1980.

Regulatory Guide (RG) 1.13, "Spent Fuel Storage Facility Design Basis," provides methods that are acceptable to the NRC staff for implementing General Design Criteria 61 of Appendix A to 10 CFR Part 50 which requires that fuel storage and handling systems be designed to assure adequate safety under normal and postulated accident conditions.

NUREG-0612 provides regulatory guidelines in two phases (Phases I and II) for licensees to assure safe handling of heavy loads in areas where a load drop could impact on stored SF, fuel in the reactor core, or equipment that may be required to achieve safe shutdown or permit continued decay heat removal. Phase I guidelines address measures for reducing the likelihood of dropping heavy loads and provide criteria for establishing safe load paths, procedures for load handling operations, training of crane operators, design, testing, inspection, and maintenance of cranes and lifting devices, and analyses of the impact of heavy load drops. Phase II guidelines address alternatives for mitigating the consequences of heavy load drops, including using either (1) a single-failure-proof crane for increased handling system reliability, or (2) electrical interlocks and mechanical stops for restricting crane travel, or (3) load drops and consequence analyses for assessing the impact of dropped loads on plant safety and operations.

Generic Letter (GL) 85-11, "Completion of Phase II of Control of Heavy Loads at Nuclear Power Plants, NUREG-0612," dated June 28, 1985, dismissed the need for licensees to implement the requirements of NUREG-0612, Phase II. However, GL 85-11 encouraged licensees to implement actions they perceived to be appropriate to provide adequate safety.

In addition, based upon specific instances of heavy load handling concerns, the NRC requested licensees, in NRC Bulletin (NRCB) 96-02, "Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor, and Over Safety-Related Equipment," to provide specific information detailing their extent of compliance with these guidelines. In response to NRCB 96-02, the licensee stated, in a letter dated July 19, 1996, that with three exceptions, the capabilities associated with handling heavy loads were in compliance with existing guidelines. These discrepancies included (1) plant activities that deviated from the descriptions provided in the FSAR such as the use of the jib crane for handling new fuel instead of the reactor building crane, (2) the use of Magnetic Particle Testing instead of Dye Penetrant (PT) as stated in the FSAR, and (3) not performing impact testing recommendations in ANSI N14.6, as referenced in NUREG-0612, to prevent brittle fracture for the special lifting devices. The FSAR has been updated for items 1 and 2 to reflect the correct descriptions and the staff finds the changes acceptable. For item 3, the licensee stated that an impact test was not required for the following reasons:

- The stress levels were lower than the threshold stress for performing impact testing; or
- The threshold temperatures for performing impact testing are less than the minimum plant design temperatures.

The staff finds that the above reasons provide an acceptable basis for not performing an impact test.

3.0 EVALUATION

The licensee determined that activities related to the storage of SF involves USQs. Specifically, the proposed cask, the associated handling activities, and the actual SFP design configuration are not explicitly described in the FSAR. Our evaluation of the USQs and the associated changes are as follows:

3.1 Description of SFP

Statements in Section 9.1.4.1, "Fuel Handling System - Design Bases," of the FSAR states: "A separate spent fuel cask loading area, isolated from the fuel storage pool eliminates the potential accident of dropping the cask and rupturing the fuel storage pool." This statement is incorrect. Both the cask loading area and the SF racks that house the SF reside within a common SFP. The cask loading area is adjacent to but separated from the SF racks by a 2-foot thick reinforced concrete wall which protects the spent fuel racks from the accidental impact from the cask.

The current FSAR description of the physical design of the SFP incorrectly indicates that the wall separating the cask loading area from the rest of the SFP will preclude rupturing the SFP if a cask is dropped into the cask loading area. The proposed change to the FSAR indicates that, the redundancy in the crane design will preclude any credible drop of a cask that could result in rupturing the SFP. The crane is expected to provide protection against a dropped cask that is equal to that provided by the wall. In other words, due to the capability of the crane a cask drop resulting in rupturing the SFP would not occur.

The Columbia Generating Station's program for the control of heavy loads was established prior to the Generic Design Criteria (GDCs), the Standard Review Plan and NUREG-0554, "Single Failure Proof Cranes for Nuclear Power Plants." Accordingly, the Columbia Generating Station was evaluated against criteria provided in Branch Technical Position (BTP) APCSB 9-1, "Overhead Handling Systems for Nuclear Power Plants," draft RG 1.104, "Overhead Crane Handling Systems for Nuclear Power Plants," and subsequently, NUREG-0612. Draft RG 1.104 was never issued as a final RG and was superseded by NUREG-0554, "Single Failure Proof Cranes for Nuclear Power Plants."

Columbia Generating Station was formerly known as WNP-2. As documented in NUREG-0892, Supplement 4, "Safety Evaluation Report Related to the Operation of WPPSS Nuclear Project No. 2," dated December 1983, WNP-2 was found to be in partial compliance with the NUREG-0612 guidelines and was requested to re-examine the qualifications of its crane as single failure proof. Specifically, in NUREG-0892, the staff raised concerns about the design of the Columbia Generating Station's reactor building crane. NRCB 96-02 requested that the licensee address its capabilities to move heavy loads. In response, the licensee in letters dated April 3 and June 30, 1997, provided information to the NRC regarding safe load paths for heavy loads and the SF loading and unloading process, and stated that, based on a detailed review, the reactor building crane satisfies the criteria in BTP APCSB 9-1, draft RG 1.104, and NUREG-0612, Appendix C. The licensee also stated that the crane is in accordance with two major codes: "Crane Manufacturers Association of America (CMAA) Specification 70," and ANSI B30.2, "Safety Code for Cranes, Derricks, and Hoists."

In the June 30, 1997, letter, the licensee presented the results of a detailed review of the crane qualifications against NUREG-0554. The results indicated that the reactor building crane satisfied NUREG-0554 with 10 exceptions. By letter dated July 26, 2001, the staff requested additional information regarding the 10 exceptions. The licensee responded by letter dated September 13, 2001, that of the 10 design exceptions, 8 are in full compliance with NUREG-0554 and 2 meet the intent of NUREG-0554. The staff has evaluated the licensee's response to the 10 exceptions:

(1) The main hoist controls do not prevent jogging as recommended in Section 6.4 of NUREG-0554.

Section 6.4 does not prohibit jogging. Section 6.4 states, "If jogging or plugging is to be used, the control circuits should include features to prevent abrupt changes in motion." The reactor building crane control circuitry contains secondary resistors and timers that prevent abrupt changes in motion. The main hoist also has a separate inching motor for precise control that can be used instead of jogging the main control. The staff finds this meets the NUREG-0554 criterion and is therefore acceptable.

(2) A 15 percent margin was not added to the loadings for wear susceptible parts as recommended in Section 2.2 of NUREG-0554.

The reason for the 15 percent margin for wear of susceptible parts is to ensure that wear to the crane will not result in the crane capacity dropping below its maximum critical load. The licensee evaluated the wear susceptible components of the main hoist and demonstrated that a 15 percent margin exists. In addition, the crane is subjected to reduced wear due to the crane being located inside. The staff finds this meets the NUREG-0554 criterion and is therefore acceptable.

(3) The maximum load (static and dynamic) exceeds 10 percent of the manufacturer's published breaking strength of each wire rope in the dual reeving system, contrary to what is recommended in Section 4.1 of NUREG-0554.

The maximum load on each wire is approximately 11 percent of the manufacturer's published breaking strength giving a 9 to 1 safety factor. The crane design compensates by limiting the wire line speed to 34.5 feet per minute (fpm) instead of the NUREG-0554 allowed 50 fpm. By reducing the wire speed the inertia forces are reduced on the wire. The staff finds the 11 percent maximum load on the wire in conjunction with the reduction in wire speed meets the intent of the NUREG-0554 criterion and is therefore acceptable.

(4) The hoist design speed is 5.5 fpm, 10 percent faster than the CMAA #70 recommended "slow" speed as required in Section 4.4 of NUREG-0554.

Each of the redundant hoists on the reactor building crane has an internal mechanical load break to prevent the lifted load from overhauling the motor and to limit the effects from a rope spin off. This internal break provides defense-in-depth and thereby compensates for the higher hoisting speed. The staff finds the 5.5 fpm along with the defense-in-depth provided by the mechanical load breaks meets the intent of the NUREG-0554 criterion and is therefore acceptable.

(5) Safe load paths and administrative controls will be used instead of interlocks as recommended in Section 6.2 of NUREG-0554.

The interlocks recommended in Section 6.2 prevent bridge and trolley movement when a fuel assembly is being hoisted free of the reactor vessel or fuel racks. The purpose of these interlocks are to prevent damage to a fuel assembly when its lateral movement is confined by the fuel racks or other assemblies in the reactor vessel. Since the reactor building crane is lifting casks and not fuel assemblies constrained in lateral movement by fuel racks these interlocks are not necessary. The staff finds this NUREG-0554 criterion does not apply to the reactor building crane and therefore the reactor building crane does not have to meet this criterion.

(6) The equalizer bar was designed to absorb the energy of a load shift should failure of one wire rope or one of the dual reeving systems occur, however, it was not designed to stop the hoisting movement during the failure as required in Section 6.3 of NUREG-0554.

The hoist movement will be stopped in the event of a failure of one rope or one of the dual reeving systems by the overload limiting device. The staff finds this meets the NUREG-0554 criterion and is therefore acceptable.

(7) Visual examination of the load block via plant procedures is performed annually instead of non-destructive examination (NDE) as recommended in Section 4.3 of NUREG-0554.

The NDE recommendation applies to forged or cast blocks, not to blocks consisting of bolted plates. The reactor building crane load block is constructed of bolted plates. Therefore an NDE is not necessary. A visual examination of the load block is performed annually to monitor the condition of the block. The staff finds this meets the NUREG-0554 criterion and is therefore acceptable.

(8) The bridge and trolley do not have variable speed controls or inching motor drives for incremental or fractional movements as recommended in Section 5.1 of NUREG-0554.

The crane is equipped with variable speed drives. The staff finds this meets the NUREG-0554 criterion and is therefore acceptable.

(9) Lugs to prevent uplifting with adequate allowable stresses for a safe shutdown earthquake (SSE) were to be replaced by the licensee so as to allow the bridge and trolley to remain in place during an SSE in accordance with Section 2.5 of NUREG-0554.

The licensee stated that the lugs were replaced so as to allow the the bridge and trolley to remain in place. The staff finds this meets the NUREG-0554 criterion and is therefore acceptable.

(10) The main hoist, trolley, and bridge motor circuits have thermal overload heater/relays which provide protection against excessive electrical current and excessive motor temperature. However, they do not have additional control devices such as internal thermocouples to sense excessive temperatures as recommended in Section 6.3 of NUREG-0554.

Section 6.3 states, "Means should be provided in the motor control circuits to sense and respond to such items as excessive electric current, excessive motor temperature, overspeed, overload and overtravel." NUREG-0554 does not require internal thermocouples to sense excessive temperatures. The staff finds that the thermal overload heater/relays which provide protection against excessive electrical current and excessive motor temperature meet the criterion of Section 6.3 of NUREG-0554 and is therefore acceptable.

NUREG-0612, Appendix C provides alternative means of satisfying the requirements of NUREG-0554 for assuring safe crane operation. Based on the staff's evaluation of the resolution of the above exceptions to the qualifications of the crane, the staff agrees with the licensee that the crane satisfies NUREG-0612, Appendix C. The single failure proof capability of the crane precludes a cask drop that may result in rupture of the SFP. Therefore, the staff finds the proposed change to the FSAR, which is based on the qualification of the crane as

being sufficiently redundant to preclude failure of any single component that could result in a dropped cask, acceptable.

3.2 Limitations on the Reactor Building Crane Travel

The FSAR states that limitations on reactor building crane travel preclude transporting the spent fuel casks over the spent fuel pool. There are no interlocks that prevent crane movement over the spent fuel cask pit loading area, which is part of the spent fuel pool. There are interlocks that prevent movement over the spent fuel racks. The proposed change is to add the statement to the FSAR that "Interlocks on the reactor building crane prevent travel over the spent fuel racks."

In Section 9.1.5 of NUREG-0892, Supplement 4, the staff concluded that the guidelines in NUREG-0612 had been met, indicating that in the event of a cask drop, any resulting consequences would be maintained below the following acceptable limits: (1) damage to spent fuel sufficient to cause a release of radioactive material to produce doses that exceed 1/4 of Part 100 limits; (2) damage to fuel and fuel storage racks does not result in a change in configuration such that the K_{eff} is larger than .095; and (3) damage to the spent fuel pool is limited so as to not result in water leakage that uncovers the fuel.

The staff believes that the proposed change to specify that crane travel is restricted from travel over spent fuel racks rather than over the spent fuel pool is less conservative. Therefore, a potential cask drop over the SFP could result in damage to the SFP and lead to water leakage that uncovers the fuel in the pool. However, the licensee states that its use of the crane will preclude a cask drop because the reactor building crane is single failure proof.

The staff finds the proposed change to be acceptable on the basis that the licensee has a highly reliable single-failure-proof crane in the load handling facilities, and therefore, the potential for a cask drop that could result in exceeding the limits discussed above would be precluded.

3.3 Safe Load Path - Movement of Spent Fuel Cask Over Safety-Related Equipment

The FSAR states that the safe load path for the cask does not involve any travel over safety-related equipment. The licensee states that because the actual load path is over safety-related electrical conduit associated with the fuel pool cooling system, the FSAR will be changed to state that "at no time while being transported does the cask pass over any safe shutdown equipment."

The staff evaluated the respective FSAR descriptions and proposed revisions and finds that the proposed change does not alter the objectives of the licensee's heavy loads program which is to assure that accidental dropping of a heavy load will be limited so as not to result in loss of required safe shutdown functions.

3.4 General Electric Cask (GE IF-300) vs. Holtec HI-STORM 100 Spent Fuel Cask System

The licensee plans to use the HI-STORM 100 spent fuel cask system instead of the General Electric (GE IF-300) cask, as analyzed in the FSAR, to remove spent fuel from the pool. The GE IF-300 cask is an NRC-licensed lead-shielded cask used to ship spent fuel from the nuclear power plant to a fuel processing plant. The HI-STORM 100 spent fuel cask is an NRC-approved fuel storage cask under 10 CFR Part 72.214. It is to be used as part of the HI-STORM 100 spent fuel cask system to facilitate onsite transfer and storage of spent fuel. There are significant physical differences between the GE IF-300 and the HI-STORM spent fuel casks. The GE IF-300 cask is 17.5 feet long x 7.5 feet in diameter and has a loaded weight of 80 tons when fully loaded with 18 BWR irradiated fuel bundles. The HI-STORM 100 spent fuel cask system consists of an overpack and a spent fuel storage cask or multipurpose canister (MPC) and is specifically designed to facilitate transfer of spent fuel stored in the MPC to an onsite storage facility such as an independent spent fuel storage installation (ISFSI). The loaded MPC is moved from the spent fuel cask loading area which is adjacent to but separated from the spent fuel pool by a weir gate. Then the MPC is moved to the ISFSI. The total bounding weight of the HI-STORM spent fuel cask system with the loaded MPC is 100 tons. A fully loaded MPC stores 24 BWR irradiated fuel bundles.

Section 9.1.4.2.1 of the FSAR states that the dry cask storage and cask handling facilities are designed based on a weight of 125 tons and the reactor building crane is designed to a rated load of 125 tons. Accordingly, the reactor building crane is designed to handle all loads below its 125-ton rated capacity. As a result, for all loads up 125 tons, the stresses on the crane should not reach the allowable working stresses. Therefore, the crane and other cask handling facilities (i.e., the lifting devices) have the design capacity to handle casks such as the HI-STORM 100 (100 tons) that are larger than the GE IF-300 (approximately 80 tons fully loaded) and not be subjected to stresses $\geq$ 125 tons.

Based on the above discussion, the staff finds the proposed change acceptable because the design of the crane and other cask handling devices enables the licensee to handle casks larger than the GE IF-300 such as the HI-STORM spent fuel cask and still meet the guidelines of NUREG-0612.

Based on the preceding discussions, the staff finds that the aforementioned considerations for the movement of heavy loads to support the proposed changes to the FSAR are acceptable. The proposed changes to the FSAR to (1) better describe the spent fuel pool; (2) limit the restriction on crane travel to over fuel in the SFP racks; and (3) to specify the load path for movement of the spent fuel cask will help to maintain safety during the cask movement operation in accordance with NUREG-0612. The licensee’s other change in the FSAR to use the HI-STORM spent fuel cask instead of the GE-IF 300 spent fuel cask is acceptable because the 125-ton reactor building single failure proof crane and other load handling facilities are designed to handle loads larger than the HI-STORM and still meet the guidelines of NUREG-0612.

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. 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 (66 FR 15918). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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

Principal Contributors: Brian Thomas
Jack Cushing

Date: October 26, 2001