May 14, 1998

Mr. Donald A. Reid
Senior Vice President, Operations
Vermont Yankee Nuclear Power Corporation
185 Old Ferry Road
Brattleboro, VT 05301

SUBJECT: ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE NO. DPR-28, VERMONT YANKEE NUCLEAR POWER STATION (TAC NO. MA1266)

Dear Mr. Reid:

The Commission has issued the enclosed Amendment No. 161 to Facility Operating License No. DPR-28, for the Vermont Yankee Nuclear Power Station in response to your application dated March 20, 1998. In your submittal, you proposed to modify your licensing basis to limit the time the containment purge and vent valves may be open to 90 hours per year.

The amendment approves the proposed changes to the Final Safety Analysis Report (FSAR), and requires that the changes be submitted with the next update of the FSAR pursuant to 10 CFR 50.71(e). The enclosed Safety Evaluation contains the staff's review and findings.

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

Sincerely,

Richard P. Croteau, Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-271

Enclosures: 1. Amendment No. 161 to License No. DPR-28
2. Safety Evaluation

cc w/encls: See next page

DISTRIBUTION
See attached

DOCUMENT NAME: G:\CROTEAU\VYMA1266.AMD

To receive a copy of this document, indicate in the box: "**C**" = Copy without attachment/enclosure "**E**" = Copy with attachment/enclosure "**N**" = No copy * see previous concurrence

OFFICE	PM:PDI-3	LA:PDI-3		OGC	D:PDI-3	
NAME	RCroteau	BClayton *		CMarco	CThomas	
DATE	05/8/98	5/4/98		05/8/98	05/14/98	

OFFICIAL RECORD COPY

9805260417 980514
PDR ADOCK 05000271
P PDR

cc:

Regional Administrator, Region I
U. S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Mr. David R. Lewis
Shaw, Pittman, Potts & Trowbridge
2300 N Street, N.W.
Washington, DC 20037-1128

Mr. Richard P. Sedano, Commissioner
Vermont Department of Public Service
120 State Street, 3rd Floor
Montpelier, VT 05602

Public Service Board
State of Vermont
120 State Street
Montpelier, VT 05602

Chairman, Board of Selectmen
Town of Vernon
P.O. Box 116
Vernon, VT 05354-0116

Mr. Richard E. McCullough
Operating Experience Coordinator
Vermont Yankee Nuclear Power Station
P.O. Box 157
Governor Hunt Road
Vernon, VT 05354

G. Dana Bisbee, Esq.
Deputy Attorney General
33 Capitol Street
Concord, NH 03301-6937

Chief, Safety Unit
Office of the Attorney General
One Ashburton Place, 19th Floor
Boston, MA 02108

Ms. Deborah B. Katz
Box 83
Shellburne Falls, MA 01370

Mr. Raymond N. McCandless
Vermont Division of Occupational
and Radiological Health
Administration Building
Montpelier, VT 05602

Mr. Gautam Sen
Licensing Manager
Vermont Yankee Nuclear Power
Corporation
185 Old Ferry Road
Brattleboro, VT 05301

Resident Inspector
Vermont Yankee Nuclear Power Station
U. S. Nuclear Regulatory Commission
P.O. Box 176
Vernon, VT 05354

Mr. Peter LaPorte, Director
ATTN: James Muckerheide
Massachusetts Emergency Management
Agency
400 Worcester Rd.
P.O. Box 1496
Framingham, MA 01701-0317

Jonathan M. Block, Esq.
Main Street
P. O. Box 566
Putney, VT 05346-0566

DATED: May 14, 1998

AMENDMENT NO. 161 TO FACILITY OPERATING LICENSE NO. DPR-28 VERMONT YANKEE NUCLEAR POWER STATION

DISTRIBUTION
Docket File
PUBLIC
J. Zwolinski
C. Thomas
R. Croteau
T. Clark
C. Berlinger
R. Lobel
OGC
G. Hill (2)
W. Beckner
ACRS
T. Harris (TLH3)
C. Cowgill, RI

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

VERMONT YANKEE NUCLEAR POWER CORPORATION

DOCKET NO. 50-271

VERMONT YANKEE NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 161
License No. DPR-28

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

   A. The application for amendment filed by the Vermont Yankee Nuclear Power Corporation (the licensee) dated March 20, 1998, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and 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.

9805260420 980514
PDR ADOCK 05000271
P PDR

2. Accordingly, the license is hereby amended to authorize revision of the Final Safety Analysis Report (FSAR) as set forth in the application for amendment by Vermont Yankee Nuclear Power Corporation dated March 20, 1998. The licensee shall submit the revised description authorized by this amendment with the next update of the FSAR in accordance with 10 CFR 50.71(e).

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Cecil O. Thomas

Cecil O. Thomas, Director
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specification

Date of Issuance: May 14, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 161

FACILITY OPERATING LICENSE NO. DPR-28

DOCKET NO. 50-271

Replace the following pages of Appendix A Technical Specification Bases with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove	Insert
165	165
165a	165a
	165b
188	188
	188a

BASES: 3.7 (Cont'd)

The vacuum relief system from the pressure suppression chamber to Reactor Building consists of two 100% vacuum relief breakers (2 parallel sets of 2 valves in series). Operation of either system will maintain the pressure differential less than 2 psig; the external design pressure is 2 psig. With one vacuum breaker out of service there is no immediate threat to accident mitigation or primary containment and, therefore, reactor operation can be continued for 7 days while repairs are being made.

The capacity of the ten (10) drywell vacuum relief valves is sized to limit the pressure differential between the suppression chamber and drywell during post-accident drywell cooling operations to the design limit of 2 psig. They are sized on the basis of the Bodega Bay pressure suppression tests. The ASME Boiler and Pressure Vessel Code, Section III, Subsection B, for this vessel allows eight (8) operable valves, therefore, with two (2) valves secured, containment integrity is not impaired.

Each drywell-suppression chamber vacuum breaker is fitted with a redundant pair of limit switches to provide fail-safe signals to panel mounted indicators in the Reactor Building and alarms in the Control Room when the disks are open more than 0.050" at all points along the seal surface of the disk. These switches are capable of transmitting the disk closed to open signal with 0.01" movement of the switch plunger. Continued reactor operation with failed components is justified because of the redundance of components and circuits and, most importantly, the accessibility of the valve lever arm and position reference external to the valve. The fail safe feature of the alarm circuits assures operator attention if a line fault occurs.

The requirement to inert the containment is based on the recommendation of the Advisory Committee on Reactor Safeguards. This recommendation, in turn, is based on the assumption that several percent of the zirconium in the core will undergo a reaction with steam during the loss-of-coolant accident. This reaction would release sufficient hydrogen to result in a flammable concentration in the primary containment building. The oxygen concentration is therefore kept below 4% to minimize the possibility of hydrogen combustion.

General Electric has estimated that less than 0.1% of the zirconium would react with steam following a loss-of-coolant due to operation of emergency core cooling equipment. This quantity of zirconium would not liberate enough hydrogen to form a combustible mixture.

The use of the 18" purge and vent flow path isolation valves AC-7A (16-19-7A), AC-7B (16-19-7B), AC-8 (16-19-8), AC-10 (16-19-10) has been restricted to 90 hours per year. Normal plant operations (other than inerting and de-inerting) will have AC-8 and AC-10 closed and nitrogen will be supplied to the drywell via the 1" nitrogen makeup supply. The differential pressure maintained between the drywell and torus will allow the nitrogen to "bubble over" into the suppression chamber. A normally open AC-6B (3") allows for venting. A normally closed AC-6A (3") is periodically opened for performance of

BASES: 3.7 (Cont'd)

surveillances such as monthly torus to drywell vacuum breaker tests. Procedurally, when AC-6A is open, AC-6 and AC-7 are closed to prevent overpressurization of the SBGT system or the reactor building ductwork, should a LOCA occur. For this and similar analyses performed, a spurious opening of AC-6 or AC-7 (one of the closed containment isolation valves) is not assumed as a failure simultaneous with a postulated LOCA. Analyses demonstrate that for normal plant operation system alignments, including surveillances such as those described above, that SBGT integrity would be maintained if a LOCA was postulated. Therefore, during normal plant operations, the 90 hour clock does not apply. Accordingly, opening of the 18 inch atmospheric control isolation valves AC-7A, AC-7B, AC-8 and AC-10 will be limited to 90 hours per calendar year (except for performance of the subject valve stroke time surveillances - in which case the appropriate corresponding valves are closed to protect equipment should a LOCA occur). This restriction will apply whenever primary containment integrity is required. The 90 hour clock will apply anytime purge and vent evolutions can not assure the integrity of the SBGT trains or related equipment.

B. and C. Standby Gas Treatment System and Secondary Containment System

The secondary containment is designed to minimize any ground level release of radioactive materials which might result from a serious accident. The Reactor Building provides secondary containment during reactor operation, when the drywell is sealed and in service; the Reactor Building provides primary containment when the reactor is shutdown and the drywell is open, as during refueling. Because the secondary containment is an integral part of the complete containment system, secondary containment is required at all times that primary containment is required except, however, for initial fuel loading and low power physics testing.

In the Cold Shutdown condition or the Refuel Mode,the probability and consequences of the LOCA are reduced due to the pressure and temperature limitations in these conditions. Therefore, maintaining Secondary Containment Integrity is not required in the Cold Shutdown condition or the Refuel Mode, except for other situations for which

significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel, during alteration of the Reactor Core, or during movement of irradiated fuel assemblies or the fuel cask in the secondary containment.

With the reactor in the Run Mode, the Startup Mode, or the Hot Shutdown condition, if Secondary Containment Integrity is not maintained, Secondary Containment Integrity must be restored within 4 hours. The 4 hours provides a period of time to correct the problem that is commensurate with the importance of maintaining secondary containment during the Run Mode, the Startup Mode, and the Hot Shutdown condition. This time period also ensures that the probability of an accident (requiring Secondary Containment Integrity) occurring during periods where Secondary Containment Integrity is not maintained, is minimal. If Secondary Containment

BASES: 3.7 (Cont'd)

Integrity cannot be restored within the required time period, the plant must be brought to a mode or condition in which the LCO does not apply.

Movement of irradiated fuel assemblies or the fuel cask in the secondary containment, alteration of the Reactor Core, and operations with the potential for draining the reactor vessel can be postulated to cause fission product release to the secondary containment. In such cases, the secondary containment is the only barrier to release of fission products to the environment. Alteration of the Reactor Core and movement of irradiated fuel assemblies and the fuel cask must be immediately suspended if Secondary Containment Integrity is not maintained. Suspension of these activities shall not preclude completing an action that involves moving a component to a safe position. Also, action must be immediately initiated to suspend operations with the potential for draining the reactor vessel to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until operations with the potential for draining the reactor vessel are suspended.

BASES: 3.8 (Cont'd)

I. Ventilation Exhaust Treatment

The requirement that the AOG Building and Radwaste Building HEPA filters be used when specified provides reasonable assurance that the release of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable". This specification implements the requirements of 10CFR Part 50.36a and the design objective of Appendix I to 10CFR Part 50. The requirements governing the use of the appropriate portions of the gaseous radwaste filter systems were specified by the NRC in NUREG-0473, Revision 2 (July 1979) as a suitable fraction of the guide set forth in Sections II.B and II.C of Appendix I, 10CFR Part 50, for gaseous effluents.

J. Explosive Gas Mixture

The hydrogen monitors are used to detect possible hydrogen buildups which could result in a possible hydrogen explosion. Automatic isolation of the off-gas flow would prevent the hydrogen explosion and possible damage to the augmented off-gas system. Maintaining the concentration of hydrogen below its flammability limit provides assurance that the releases of radioactive materials will be controlled.

K. Steam Jet Air Ejector (SJAE)

Restricting the gross radioactivity release rate of gases from the main condenser SJAE provides reasonable assurance that the total body exposure to an individual at the exclusion area boundary will not exceed a small fraction of the limits of 10CFR Part 100 in the event this effluent is inadvertently discharged directly to the environment without treatment. This specification implements the requirements of General Design Criteria 60 and 64 of Appendix A to 10CFR Part 50.

L. Primary Containment (MARK I)

This specification provides reasonable assurance that releases from containment purging/venting operations will be filtered through the Standby Gas Treatment System (SBGT) so that the annual dose limits of 10CFR Part 20 for Members of the Public in areas at and beyond the Site Boundary will not be exceeded. The dose objectives of Specification 3.8.G restrict purge/venting operations when the Standby Gas Treatment System is not in use and gives reasonable assurance that all releases from the plant will be kept "as low as is reasonably achievable". The specification requires the use of SBGT only when Iodine-131, Iodine-133 or radionuclides in particulate form with half-lives greater than 8 days in containment exceeds the levels in Table 1, Column 3, to Appendix B of 10CFR 20.1001-20.2401 since the filter system is not considered effective in reducing noble gas radioactivity from gas streams.

The use of the 18" purge and vent flow path isolation valves AC-7A (16-19-7A), AC-7B (16-19-7B), AC-8 (16-19-8), AC-10 (16-19-10) has been restricted to 90 hours per year. Normal plant operations (other than inerting and de-inerting) will have AC-8 and AC-10 closed and nitrogen will be supplied to the drywell via the 1" nitrogen makeup supply. The differential pressure maintained between the drywell

BASES: 3.8 (Cont'd)

and torus will allow the nitrogen to "bubble over" into the suppression chamber. A normally open AC-6B (3") allows for venting. A normally closed AC-6A (3") is periodically opened for performance of surveillances such as monthly torus to drywell vacuum breaker tests. Procedurally, when AC-6A is open, AC-6 and AC-7 are closed to prevent overpressurization of the SBGT system or the reactor building ductwork, should a LOCA occur. For this and similar analyses performed, a spurious opening of AC-6 or AC-7 (one of the closed containment isolation valves) is not assumed as a failure simultaneous with a postulated LOCA. Analyses demonstrate that for normal plant operation system alignments, including surveillances such as those described above, that SBGT integrity would be maintained if a LOCA was postulated. Therefore, during normal plant operations, the 90 hour clock does not apply. Accordingly, opening of the 18 inch atmospheric control isolation valves AC-7A, AC-7B, AC-8 and AC-10 will be limited to 90 hours per calendar year (except for performance of the subject valve stroke time surveillances - in which case the appropriate corresponding valves are closed to protect equipment should a LOCA occur). This restriction will apply whenever primary containment integrity is required. The 90 hour clock will apply anytime purge and vent evolutions can not assure the integrity of the SBGT trains or related equipment.

M. Total Dose (40CFR190)

This specification is provided to meet the dose limitations of 40CFR Part 190 to Members of the Public in areas at and beyond the Site Boundary. The specification requires the preparation and submittal of a Specific Report whenever the calculated doses from plant radioactive effluents exceed twice the design objective doses of Appendix I. For sites containing up to 4 reactors, it is highly unlikely that the resultant dose to a Member of the Public will exceed the dose limits of 40CFR Part 190 if the individual reactors remain within the reporting requirement level. The Special Report will describe a course of action that should result in the limitation

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 161 TO FACILITY OPERATING LICENSE NO. DPR-28

VERMONT YANKEE NUCLEAR POWER CORPORATION

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

1.0 INTRODUCTION

The Vermont Yankee Nuclear Power Station is a boiling water reactor (BWR), model BWR-4, with a Mark I containment. By letter dated March 20, 1998, the Vermont Yankee Nuclear Power Corporation, the licensee for the Vermont Yankee Nuclear Power Plant, submitted for Nuclear Regulatory Commission (NRC) staff review a proposed change to the licensing bases described in the Final Safety Analysis Report (FSAR). This change involves a licensee-imposed limit on opening certain primary containment isolation valves. The valves would be open for no more than 90 hours per calendar year in order to minimize the possibility of overpressurizing the standby gas treatment system (SGTS) if a large break LOCA should occur while purging the primary containment.

Vermont Yankee Licensee Event Report (LER) 97-005-01, dated September 5, 1997 reported that the potential existed at Vermont Yankee Nuclear Power Station to overpressurize the standby gas treatment system (SGTS) filter train housings during inerting or deinerting operations at power in the event of a loss-of-coolant accident (LOCA). Although the SGTS is designed to isolate automatically on a primary containment isolation signal, the stroke times of the isolation valves were such that overpressurization of the SGTS was predicted. The licensee has established administrative controls to prohibit the system alignments which could result in overpressurizing the SGTS, given a LOCA. However, these restrictions reduce operational flexibility. The licensee points out in the March 20, 1998 submittal that there are important advantages to maintaining the capability to vent and purge the containment during modes of operation in which containment isolation is required. Providing the capability to purge and vent (inert and deinert) the primary containment during these periods enables personnel entry for maintenance and inspection of equipment and the capability to verify that "all RCS boundaries that have been interrupted during the refueling outage have been returned to an operable condition."

In accordance with existing NRC guidance (Standard Review Plan Section 6.2.4, "Containment Isolation System"), the licensee proposed to limit vent and purge operations to no more than 90 hours per year. This limit would be located in the technical specifications bases and FSAR.

9805260421 980514
PDR ADOCK 05000271
P PDR

2.0 EVALUATION

Vermont Yankee LER 97-005-01 reports that on March 24, 1997, the licensee determined that the potential existed to overpressurize the SGTS filter train housings, should a LOCA occur while the primary containment was being inerted or deinerted using the SGTS. Amendment 91, dated October 28, 1985, specifically included a consideration of a LOCA occurring during inerting and deinerting activities in the licensing basis. The licensee performed a radiological dose calculation which demonstrated that all regulatory limits would be satisfied given this event.

However, although the SGTS is designed to isolate automatically from the primary containment on a LOCA signal, LER 97-005-01 reported that the licensee determined that the valve stroke times were such that there was a potential for SGTS overpressurization because the LOCA may generate a pressure pulse of such a magnitude and velocity that the isolation valve could not prevent overpressurization of the SGTS. The design pressure of the SGTS housing is 2 psig.

Vermont Yankee technical specifications permit inerting and deinerting of the primary containment for no more than 24 hours following startup and prior to a shutdown. Thus, the time period during which this situation could occur is limited. The licensee put in place administrative controls to prohibit opening of the torus and drywell 18-inch valves.

Further analyses by the licensee, as discussed in LER 97-014-0, Revision 1, showed that over-pressurization of the SGTS housings could also occur during purging and venting activities utilizing a smaller (3-inch) drywell vent bypass valve (AC-6A).

In order to regain operational flexibility to inert and deinert the primary containment at power, the licensee proposes to limit the allowable time during a calendar year that these operations can be performed to 90 hours.

An NRC letter to the Yankee Atomic Electric Company, referring to docket 50-271 (the Vermont Yankee Nuclear Power Station), dated November 29, 1978, provided guidance on the use of containment purge and vent systems during modes of operation in which containment integrity is required. This letter discussed operating reactor problems with purge and vent operations that could compromise containment integrity and specified actions licensees should take to avoid these problems. For licensees proposing limited purging of containment, as an alternative to a radiological dose analysis, this guidance limited purging and venting to 90 hours per calendar year, as prescribed in Standard Review Plan Section (SRP) 6.2.4, Containment Isolation System and Branch Technical Position 6-4, Containment Purging During Normal Operations. By limiting use of the purge and vent paths to 90 hours per calendar year, the licensee is complying with this guidance.

The November 29, 1978, letter also requested that justification for limited purging include a demonstration (by test or analysis similar to that specified in SRP Section 3.9.3) of the ability of the purge and vent valves to close under postulated accident conditions. In a May 22, 1984, NRC letter to the Vermont Yankee Nuclear Power Corporation, the staff concluded that the

licensee had demonstrated the ability of Vermont Yankee's containment purge and vent valves to close against the buildup of containment pressure in the event of a design basis LOCA. The conclusion was based on valves V16-19-7 and V16-19-7a being blocked at 50 degrees open. This restriction on opening is specified in Vermont Yankee technical specification Table 4.7.2a and discussed in Basis section 4.7.D. The licensee has not proposed changing these technical specifications.

The November 29, 1978, letter also requested that the 90-hour limit on purging be included in the plant's technical specifications. The licensee has proposed instead that the 90-hour limit be placed in the technical specifications bases (3.7.A and 3.8.L) and FSAR. The staff finds this an acceptable location for the limit. The Improved Standard Technical Specifications for BWR-4s, NUREG-1433, Revision 1, do not specify a time limit for purging operations. Rather, permissible reasons for purging are specified.

In the NRC staff's Final Safety Evaluation Report for the Advanced BWR (ABWR) design (NUREG-1503, Volume 1, July 1994) the staff stated that an analysis demonstrating that the use of the SGTS during inerting, deinerting, pressure control or purging of the primary containment during normal plant operation will not impair its functional capability during a design basis accident was not necessary provided the use of the SGTS during power operation is limited to no more than 90 hours per year. No technical specifications were needed to address this limit.

Therefore, the staff finds the licensee's proposal to include a limit in the FSAR (and technical specifications bases) to be consistent with staff guidance and positions with respect to technical specifications and is, therefore, acceptable.

As noted above, Vermont Yankee LER 97-014-01 pointed out that not only could the SGTS be damaged by a pressure pulse and flow through the 18-inch purge valves, but also by flow through the drywell 3-inch vent valve. The bases state that when this valve (AC-6A) is open, other valves will be closed by procedure to prevent overpressurization of the SGTS or the reactor building ductwork should a LOCA occur. The staff finds this to be acceptable since the procedural controls will keep SGTS isolated when the drywell 3-inch vent valve is open.

In summary, the licensee has taken appropriate steps to ensure that the risk of a large LOCA with concomitant overpressurization and failure of the standby gas treatment system is acceptably low. The licensee used previous NRC staff guidance to limit the time the containment purge and vent valves may be open. Therefore, the NRC staff finds the licensee's proposal acceptable.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to 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 (63 FR 14976). 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.

5.0 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: Richard Lobel

Date: May 14, 1998