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

April 14, 2000

Mr. James Knubel
Chief Nuclear Officer
Power Authority of the State of New York
123 Main Street
White Plains, NY 10601

SUBJECT: JAMES A. FITZPATRICK NUCLEAR POWER PLANT - ISSUANCE OF AMENDMENT RE: CHANGES TO THE TECHNICAL SPECIFICATIONS REGARDING THE ALLOWED CONTAINMENT LEAKAGE RATE (TAC NO. MA1136)

Dear Mr. Knubel:

The Commission has issued the enclosed Amendment No. 261 to Facility Operating License No. DPR-59 for the James A. FitzPatrick Nuclear Power Plant (JAFNPP). The amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated February 26, 1998, as supplemented October 14, 1999.

The amendment changes the TS by changing the value of the allowable containment leakage rate to 1.5 percent per day and correcting conflicting information in TS Section 4.6.C, "Coolant Chemistry." However, the acceptance of this amendment does not relieve you from responding to the U.S. Nuclear Regulatory Commission (NRC) Generic Letter 99-02 or regulatory actions that may be proposed in the future as the NRC-Nuclear Energy Institute task force resolves the control room habitability generic issues.

A copy of the related safety evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

Guy S. Vissing

Guy S. Vissing, Sr. Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-333

Enclosures: 1. Amendment No. 261 to DPR-59
2. Safety Evaluation

cc w/encls: See next page

ATTACHMENT TO LICENSE AMENDMENT NO. 261

FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

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

Remove Pages	Insert Pages
139	139
140	140
191	191
193	193
258e	258e
285	285
285a	285a

Replace the following pages of the Appendix B Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change

Remove Page	Insert Page
1	1

3.6 (cont'd)

B. Deleted

C. Coolant Chemistry

1. The reactor coolant system radioactivity concentration in water shall not exceed the equilibrium value of 0.2 µCi/gm of dose equivalent I-131. This limit may be exceeded, following a power transient, for a maximum of 48 hours. During this iodine activity transient the iodine concentrations shall not exceed the equilibrium limits by more than a factor of 10 whenever the main steamline isolation valves are open. The reactor shall not be operated more than 5 percent of its annual power operation under this exception to the equilibrium limits. If the iodine concentration exceeds the equilibrium limit by more than a factor of 10, the reactor shall be placed in a cold condition within 24 hours.

4.6 (cont'd)

B. Deleted

C. Coolant Chemistry

1. a. A sample of reactor coolant shall be taken at least every 96 hours and analyzed for gross gamma activity.

   b. Isotopic analysis of a sample of reactor coolant shall be made at least once/month.

   c. A sample of reactor coolant shall be taken prior to startup and at 4 hour intervals during startup and analyzed for gross gamma activity.

   d. During plant steady state operation and following an offgas activity increase (at the Steam Jet Air Ejectors) of 10,000 µCi/sec within a 48 hour period or a power level change of ≥20 percent of full rated power/hr reactor coolant samples shall be taken and analyzed for gross gamma activity. At least three samples will be taken at 4 hour intervals. These sampling requirements may be omitted whenever the equilibrium I-131 concentration in the reactor coolant is less than 0.002 µCi/ml.

4.6 (cont'd)

e. If the gross activity counts made in accordance with a, c, and d above indicate a total iodine concentration in excess of 0.002 μCi/ml, a quantative determination shall be made for I-131 and I-133.

2. The reactor coolant water shall not exceed the following limits with steaming rates less than 100,000 lb/hr except as specified in 3.6.C.3:

   Conductivity 2 μmho/cm
   Chloride ion 0.1 ppm

3. For reactor startups the maximum value for conductivity shall not exceed 10 μmho/cm and the maximum value for chloride ion concentration shall not exceed 0.1ppm, for the first 24 hours after placing the reactor in the power operating condition. During reactor shutdowns, specification 3.6.C.4 will apply.

2. During startups and at steaming rates below 100,000 lb/hr, and when the conductivity of the reactor coolant exceeds 2 μmhos/cm, a sample of reactor coolant shall be taken every 4 hr and analyzed for conductivity and chloride content.

3. a. With steaming rates greater than or equal to 100,000 lb/hr, a reactor coolant sample shall be taken at least every 96 hours and whenever the continuous conductivity monitors indicate abnormal conductivity (other than short-term spikes), and analyzed for conductivity and chloride ion content.

   b. When the continuous conductivity monitor is inoperable, a reactor coolant sample shall be taken at least daily and analyzed for conductivity and chloride ion content.

3.7 BASES (cont'd)

complete containment system, secondary containment is required at all times that primary containment is required as well as during refueling.

The Standby Gas Treatment System is designed to filter and exhaust the reactor building atmosphere to the main stack during secondary containment isolation conditions with a minimum release of radioactive materials from the reactor building to the environs. Both standby gas treatment fans are designed to automatically start upon containment isolation; however, only one fan is required to maintain the reactor building pressure at approximately a negative 1/4 in. water gage pressure; all leakage should be in-leakage. Each of the two fans has 100 percent capacity. If one Standby Gas Treatment System circuit is inoperable, the other circuit must be verified operable daily. This substantiates the availability of the operable circuit and results in no added risk; thus, reactor operation or refueling operation can continue. If neither circuit is operable, the Plant is brought to a condition where the system is not required.

While only a small amount of particulates is released from the Pressure Suppression Chamber System as a result of the loss-of-coolant accident, high-efficiency particulate filters are specified to minimize potential particulate release to the environment and to prevent clogging of the iodine filter. The high-efficiency filters have an efficiency greater than 99 percent for particulate matter larger than 0.3 micron. The minimum iodine removal efficiency is 99 percent. Filter banks will be replaced whenever significant changes in filter efficiency occur. Tests (11) of impregnated charcoal identical to that used in the filters indicated that shelf life up to 5 yr leads to only minor decreases in methyl iodine removal efficiency.

The analysis of the design basis loss-of-coolant accident assumed a charcoal filter efficiency of 90% for the SBGT system and a source term provided by GE based on NEDO-10871. The assumed 90% is sufficient to prevent exceeding 10CFR100 guidelines for accidents analyzed. The charcoal and particulate filters are tested to an acceptance criteria of 99% efficiency with 1% penetration. A heater maintains relative humidity below 70% in order to assure the efficient removal of methyl iodine on the impregnated charcoal filters. Regulatory Guide 1.52 assigns a charcoal filter efficiency of 95% for 2 inch beds (as used in the SBGT system), thus assuming 90% efficiency in dose calculations and testing to 99% efficiency provide additional conservatism in analysis and operation.

The operability of the Standby Gas Treatment System (SGTS) must be assured if a design basis loss of coolant accident (LOCA) occurs while the containment is being purged or vented through the SGTS. Flow from containment to the SGTS is via 6 inch Valve Number 27MOV-121. Since the maximum flow through the 6 inch line following a design basis LOCA is within the design capabilities of the SGTS, use of the 6 inch line assures the operability of the SGTS.

D. <u>Primary Containment Isolation Valves</u>

Double isolation valves are provided on lines penetrating the primary containment and open to the free space

4.7 BASES

A. Primary Containment

The water in the suppression chamber is used only for cooling in the event of an accident; i.e., it is not used for normal operation; therefore, a daily check of the temperature and volume is adequate to assure that adequate heat removal capability is present.

The primary containment preoperational test pressures are based upon the calculated primary containment pressure response corresponding to the design basis loss-of-coolant accident. The peak drywell pressure would be about 45 psig which would rapidly reduce to 27 psig within 30 sec. following the pipe break. Following the pipe break, the suppression chamber pressure rises to 26 psig within 30 sec, equalizes with drywell pressure and thereafter rapidly decays with the drywell pressure decay (14).

The design pressure of the drywell and suppression chamber is 56 psig(15). The design basis accident leakage rate is 1.5 percent/day at a pressure of 45 psig. As pointed out above, the drywell and suppression chamber pressure following an accident would equalize fairly rapidly. Based on the primary containment pressure response and the fact that the drywell and suppression chamber function as a unit, the primary containment will be tested as a unit rather than the individual components separately.

Design basis accidents were evaluated as discussed in Section 14.6 of the FSAR and the power uprate safety evaluation, Reference 18. The whole body and thyroid doses in the control room, low population zone (LPZ) and site boundary meet the requirements of 10 CFR Parts 50 and 100. The technical support center (TSC), not designed to these licensing bases, was also analyzed. The whole body and thyroid dose acceptance criteria used for the main control room are met for the TSC when initial access to the TSC and occupancy of certain areas in the TSC is restricted by administrative control. The LOCA dose evaluations, References 19, 20, and 21, assumed: the primary containment leak rate was 1.5 volume percent per day; source term releases were in accordance with TID-14844 and Regulatory Guide 1.3, and were consistent with the Standard Review Plan; and the standby gas treatment system filter efficiency was 90% for halogens. These doses are also based on the

6.19 POSTACCIDENT SAMPLING PROGRAM

A program shall be established, implemented, and maintained which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

A) Training of personnel,

B) Procedures for sampling and analysis,

C) Provisions for maintenance of sampling and analysis

6.20 PRIMARY CONTAINMENT LEAKAGE RATE TESTING PROGRAM

A program shall be established to implement the leakage rate testing of the Primary Containment as required by 10 CFR 50.54 (o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program", dated September 1995, as modified by the exception that Type C testing of valves not isolable from the containment free air space may be accomplished by pressurization in the reverse direction provided that testing in this manner provides equivalent or more conservative results than testing in the accident direction. If potential atmospheric leakage paths (e.g., valve stem packing) are not subjected to test pressure, the portions of the valve not exposed to test pressure shall be subjected to leakage rate measurement during regularly scheduled Type A testing. A list of these valves, the leakage rate measurement method, and the acceptance criteria, shall be contained in the Program.

A. The peak Primary Containment internal pressure for the design basis loss of coolant accident (P_a), is 45 psig.

B. The maximum allowable Primary Containment leakage rate (L_a), at P_a, shall be 1.5% of primary containment air weight per day.

C. The leakage rate acceptance criteria are:

1. Primary containment leakage rate acceptance criteria is $\leq$ 1.0 L_a. During unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq$ 0.60 L_a for the Type B and Type C tests and $\leq$ 0.75 L_a for the Type A tests;

2. Airlock testing acceptance criteria are:

   a. Overall airlock leakage rate is $\leq$ 0.05 L_a when tested at $\geq$ P_a,

   b. For each door seal, leakage rate is $\leq$ 120 scfd when tested at $\geq$ P_a.

3. MSIV leakage rate acceptance criteria is $\leq$ 11.5 scfh for each MSIV when tested at $\geq$ 25 psig.

D. The provisions of Specification 4.0.B do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

E. The provisions of Specification 4.0.C are applicable to the Primary Containment Leakage Rate Testing Program.

7.0 REFERENCES

(1) E. Janssen, "Multi-Rod Burnout at Low Pressure," ASME Paper 62-HT-26, August 1962.

(2) K.M. Backer, "Burnout Conditions for Flow of Boiling Water in Vertical Rod Clusters," AE-74 (Stockholm, Sweden), May 1962.

(3) FSAR Section 11.2.2.

(4) FSAR Section 4.4.3.

(5) I.M. Jacobs, "Reliability of Engineered Safety Features as a Function of Testing Frequency," Nuclear Safety, Vol. 9, No. 4, July-August 1968, pp 310-312.

(6) Deleted

(7) I.M. Jacobs and P.W. Mariott, APED Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards - April 1969.

(8) Bodega Bay Preliminary Hazards Report, Appendix 1, Docket 50-205, December 28, 1962.

(9) C.H. Robbins, "Tests of a Full Scale 1/48 Segment of the Humbolt Bay Pressure Suppression Containment," GEAP-3596, November 17, 1960.

(10) "Nuclear Safety Program Annual Progress Report for Period Ending December 31, 1966, ORNL-4071."

(11) Section 5.2 of the FSAR.

(12) TID 20583, "Leakage Characteristics of Steel Containment Vessel and the Analysis of Leakage Rate Determinations."

(13) Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program", dated September 1995.

(14) Section 14.6 of the FSAR.

(15) ASME Boiler and Pressure Vessel Code, Nuclear Vessels, Section III. Maximum allowable internal pressure is 62 psig.

(16) 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors, Option B - Performance Based Requirements", Effective Date October 26, 1995

(17) Deleted

(18) General Electric Report NEDC-32016P-1, "Power Uprate Safety Analysis for James A. FitzPatrick Nuclear Power Plant," April 1993 (proprietary), Including Errata and Addenda Sheet No. 1, dated January 1994.

(19) James A. FitzPatrick Calculation JAF-CALC-RAD-00023, Rev. 1, "Power Uprate Program - Technical Support Center Post-Accident Radiological Habitability Study," September 1999.

(20) James A. FitzPatrick Calculation JAF-CALC-RAD-00042, Rev. 2, "Control Room Radiological Habitability Under Power Uprate Conditions and CREVASS Reconfiguration," October 1997.

7.0 REFERENCES (continued)

(21) James A. FitzPatrick Calculation JAF-CALC-RAD-00048, Rev. 1, "Power Uprate Project - Radiological Impact at Onsite and Offsite Outdoor Receptors Following Design-Basis Accidents," November 1997

(22) General Electric Report GE-NE-187-45-1191, "Containment Systems Evaluation for the James A. FitzPatrick Nuclear Power Plant," November 1991 (proprietary).

(23) James A. FitzPatrick Calculation JAF-CALC-RAD-00007, Rev. 2, "Power Uprate Program - Onsite and Offsite Post-Accident Atmosphere Dispersion Factors", August 1997.

RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATIONS

1.0 DEFINITIONS

A. Dose Equivalent I-131
The Dose Equivalent I-131 is the concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in International Commission on Radiological Protection Publication 30 (ICRP-30), "Limits for Intake by Workers" or in NRC Regulatory Guide 1.109, Revision 1, October 1977.

B. Instrument Channel Calibration
See Appendix A Technical Specifications.

C. Instrument Channel Functional Test
See Appendix A Technical Specifications.

D. Instrument Check
See Appendix A Technical Specifications.

E. Logic System Function Test
See Appendix A Technical Specifications.

F. Member(s) of the Public
Member(s) of the Public includes all persons who are not occupationally associated with the facilities on the NYPA/(NMPC) Niagara Mohawk Power Corporation site. This category does not include employees of the utilities, its contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plants.

G. Offgas Treatment System
The Offgas Treatment System is the system designed and installed to: reduce radioactive gaseous effluents by collecting primary coolant system offgases from the main condenser; and, providing for delay of the offgas for the purpose of reducing the total radioactivity prior to release to the environment.

H. Offsite Dose Calculation Manual (ODCM)
The ODCM describes the methodology and parameters to be used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and in the calculation of gaseous and liquid effluents monitoring instrumentation alarm/trip set points and in the conduct of the environmental monitoring program.

I. Operable
See Appendix A Technical Specifications.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 261 TO FACILITY OPERATING LICENSE NO. DPR-59

POWER AUTHORITY OF THE STATE OF NEW YORK

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

DOCKET NO. 50-333

1.0 INTRODUCTION

By letter dated February 26, 1998, as supplemented by letter dated October 14, 1999, the Power Authority of the State of New York (PASNY or the licensee), the licensee for James A. FitzPatrick Nuclear Power Plant, requested an amendment to the technical specifications (TSs) for the FitzPatrick Nuclear Power Plant requesting an increase in the maximum allowable primary containment leakage rate. The October 14, 1999, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The proposed amendment would (1) increase the allowable containment leakage rate (L_a) to 1.5 weight percent (w/o) per day from 0.5 w/o per day.

Specifically, the licensee requested the following changes to the FitzPatrick TSs.

- the maximum allowable primary containment leakage rate (L_a) limit in TS Section 6.20 (page 258e) and TS Bases Section 4.7 (page 193) be amended to 1.5 w/o per day from 0.5 w/o per day.
- the standby gas treatment system (SGTS) charcoal filter iodine removal efficiency in TS Bases Sections 3.7 and 4.7 (pages 191 and 193) be amended to 90 percent from 99 percent.
- the primary coolant sampling requirement threshold value in TS Section 4.6.C (pages 139 and 140) be amended to 0.002 µCi/ml from 0.007 µCi/ml of dose-equivalent iodine-131 in the primary coolant in order to have the sampling requirement omitted.

2.0 EVALUATION

To demonstrate the adequacy of the FitzPatrick engineered safety feature (ESF) systems designed to mitigate the radiological consequences of the design basis accidents (DBAs) with the increased containment leakage rate of 1.5 w/o per day, the licensee reevaluated the offsite and control room radiological consequences resulting from the postulated loss-of-coolant accident (LOCA). The licensee submitted the results of its offsite and control room radiological

consequence analyses. In its submittal, the licensee concluded that the existing ESF systems at FitzPatrick with the increased containment leakage rate of 1.5 w/o per day will provide assurance that the radiological consequences at the exclusion area boundary (EAB) and the low population zone (LPZ) resulting from a postulated LOCA will be within the dose reference values given in 10 CFR Part 100 and that the radiological consequence to the control room operator will be within the dose criteria specified in General Design Criterion 19 of 10 CFR Part 50.

To review the licensee's radiological consequence analyses, the staff performed confirmatory radiological consequence calculations for the following sources and radioactivity transport pathways to the environment after the postulated LOCA:

- Primary containment and main steam isolation valve leakage
- Post-LOCA leakage from engineered safety features systems outside containment.

In its calculation of the radiological consequences of the postulated LOCA, the staff used the source term assumptions given in Regulatory Guide (RG) 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling-Water Reactors," (Revision 2), and an NRC computer code, HABIT (Version 1.1). The computer code is described in NUREG/CR-6210 (Supplement 1) and calculates, among other things, the radiological consequence doses at the EAB and the LPZ and in the control room after design basis accidents. The primary containment was assumed to leak to the reactor building from the drywell at a constant rate of 1.5 w/o per day (including MSIV leakage) for the entire duration of the accident (30 days). The fission products leaked to the reactor building are released directly to the environment through the SGTS and the main stack without holdup or mixing in the reactor building.

The FitzPatrick SGTS consists of two safety-related, full-capacity, redundant trains. Each train has one 2-inch-deep charcoal adsorber. The licensee proposed, and the staff accepted and used in its dose calculations, a charcoal adsorber with an iodine removal efficiency of 90 percent.

Any leakage water from ESF components located outside the drywell releases fission products during the recirculating phase of long-term core cooling after the postulated LOCA. The licensee estimated this leakage to be less than 5 gallons per minute (gpm), and the staff used 5 gpm for the entire duration of the accident. The staff assumed 50 percent of the core iodine inventory is uniformly mixed with the primary coolant circulated through the drywell external piping systems and assumed 10 percent of the iodine in the liquid leakage from the external piping systems will become airborne. Because the FitzPatrick reactor building is provided with an ESF-grade filtration system to filter the reactor building exhaust, the staff has not calculated the contribution to the LOCA doses from a passive failure in an ESF component. The staff assumed that airborne fission products from the leakage are directly released to the environment through the SGTS and the main stack without holdup or mixing in the reactor building. The major parameters and assumptions used are given in Table 1, and the resulting site boundary doses are given in Table 2.

The FitzPatrick control room emergency ventilation air supply system (CREVASS) consists of two safety-related, seismic Category 1, full-capacity, redundant trains. Each train has two, 2-

inch-deep charcoal adsorbers in series. The licensee proposed, and the staff accepted and used, a charcoal adsorber iodine removal efficiency of 90 percent. Upon receipt of a high-radiation signal from the radiation monitor in the control room air intake duct, the CREVASS is manually placed in the isolation mode. There is no automatic isolation capability for the CREVASS after a postulated DBA. The licensee proposed, and the staff accepted and used in its evaluation, a 30-minute operator action time (delay) to isolate the CREVASS.

The staff used in its control room dose calculation a maximum unfiltered air infiltration rate of 15,010 standard cubic feet per minute (scfm) before isolation of the control room, based on the maximum air intake during the normal operational mode of the control room ventilation system. After isolation, the staff used a maximum unfiltered air infiltration rate of 2,110 scfm. This rate is based on a single failure of the motor-operated air intake valve to close and a failure to close a manually operated air intake damper. The filtered make-up air intake to the control room is 1,000 scfm. The major parameters and assumptions used are given in Table 1 (Attachment 1), and the resulting control room doses are given in Table 2 (Attachment 2).

2.1 Atmospheric Relative Concentrations in the Control Room and at the Exclusion Area Boundary and the Low-Population Zone

Although the licensee's submittals contain calculations of atmospheric relative concentration (X/Q) values for other design basis accidents, this safety evaluation is limited to the X/Q assessment for releases from the plant main stack associated with the postulated LOCA. The licensee's X/Q assessments for other design basis accidents and postulated release locations have not been evaluated by the staff in this review because they are not germane to this license amendment request. Thus, the staff's findings of acceptability apply only to a postulated release from the stack associated with a design basis LOCA.

The licensee provided 8 years of meteorological data, 1985 through 1992, for the Nine Mile Point Nuclear Station, which is adjacent to the FitzPatrick plant. The staff finds the use of this data for FitzPatrick appropriate for this license amendment evaluation. Data recovery rates for all years exceeded the guideline set forth in RG 1.23, "Onsite Meteorological Programs." This guide also states that the interval of measurement between lower and higher measurement levels should be at least 30 meters. The data were measured at 9.1, 30.5, and 61 meters. Although guidance would recommend use of measurements at the 9.1 and 61 meter levels, the staff's review of the measured stabilities and wind speeds clearly indicates that the meteorological tower frequently extends through a thermal internal boundary layer (TIBL) associated with lake breeze flow at Lake Erie. The licensee's submittal includes the results of a study of the TIBL and a discussion of its relevance to the dispersion and the estimation of stability classes. The X/Q values contained in the submittal are based on stability classes estimated using vertical temperature difference measurements (ΔT) between 9.1 and 30.5 meters ($\Delta T_{9\text{-}30}$).

The rationale for using the $\Delta T_{9\text{-}30}$ was provided by the licensee in a table that compares frequency distributions of stability class estimates using $\Delta T_{9\text{-}30}$, $\Delta T_{30\text{-}61}$, and $\Delta T_{9\text{-}61}$ with the distribution of stability classes estimated objectively from solar radiation, wind speed, and other related meteorological parameters. On the basis, at least in part, of this information, the licensee has elected to characterize stability for the dose calculations on the temperature difference between 9.1 and 30.5 meters, rather than between 9.1 and 61 meters. The 95th percentile X/Q values for elevated releases are associated with unstable conditions. The $\Delta T_{9\text{-}30}$

gives a higher frequency of unstable conditions than $\Delta T_{9\text{-}61}$. Consequently, the staff finds that the use of the $\Delta T_{9\text{-}30}$ interval is acceptable.

In its submittal, the licensee presented revised X/Q values used in the analysis describing the procedures and rationale upon which the revised X/Q values are based. The licensee compared the revised X/Q values with those in the FitzPatrick UFSAR and the staff's earlier safety evaluation. The submital does not include detailed descriptions of the computer codes used to calculate the X/Q values. In general, the approach used by the licensee when estimating the revised X/Q values was based on regulatory guidance provided in RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants." Additional guidance was drawn from RG 1.3, RG 1.23, RG 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," ANSI/ANS-2.5, and the XOQDOQ computer code described in NUREG/CR-2919, "XOQDOQ: Computer Program for the Meteorological Evaluation of Routine Releases at Nuclear Power Stations."

Modifications to procedures described in the guidance were based on consideration of the site geography and topography and analysis of 8 years of onsite meteorological data. The procedure modifications decrease the conservative nature of the assumptions and model in the guidance. However, the modified procedures should still be conservative because the modifications are consistent with local conditions and tend to understate the probable effects of the local conditions on reducing X/Q values. Results of confirmatory calculations, based on information in the submittal are consistent with the revised X/Q values for the 0-to 2-hour period at the site boundary and the 0-to 4-hour period at the LPZ. The staff also made confirmatory calculations for the LPZ for periods after the initial 4 hours using the onsite data. For these calculations, the peak offsite X/Q value was assumed to occur 4 miles south of the stack on elevated terrain. Because of the elevated terrain, the effective stack height was assumed to be 50 meters. The results of these calculations indicate that the X/Q values submitted by the licensee are conservative. Therefore, the staff finds that the revised X/Q values are acceptable.

For purposes of dispersion calculations, FitzPatrick is a coastal site. RG 1.145 suggests that fumigation be assumed offsite for the first 4 hours of an elevated release. The fumigation occurs when the plume intersects the TIBL. The height of the TIBL is determined by the distance from the coast, while the dispersion is a function of the distance from the release point. Consequently, fumigation X/Q values are a function of both distance from the coast and the stack. If the stack height is less than the TIBL, fumigation due to intersection of the plume and the TIBL will not occur. However, there may still be fumigation as the neutral or unstable boundary layer grows due to surface heating during morning hours. It does not appear that the licensee considered this type of fumigation. This omission is not likely to significantly affect X/Q values at the FitzPatrick site boundary and LPZ because the site X/Q values are based on the fumigation model.

The licensee did not consider fumigation in calculating X/Q values at the control room air intake for stack releases. The rationale provided for not considering fumigation is that TIBL-related fumigation would not occur for wind directions carrying radioactivity from the stack toward the control room air intake. It should also be noted that the distance between the stack and the control room air intake is sufficiently small such that if normal fumigation occurred during southerly winds, the turbulent processes that bring material to the ground could not bring stack

effluent to the ground before the material was carried past the intake. Consequently, the decision not to consider fumigation in determining X/Q values at the control room and technical support center air intakes is appropriate.

Given the stack height and the distance between the stack and the control room air intake, typical calculations of X/Q values for an elevated release using the straight-line model with standard dispersion coefficients would result in unrealistically low X/Q values for use in dose assessments for design-basis accidents. Under unstable and light wind atmospheric conditions, meteorological factors, such as plume meander and looping that are not adequately treated by the straight-line model are likely to result in the highest X/Q values. The licensee has attempted to estimate these X/Q values by assuming that the distance traveled by the plume between the stack and the control room is equal to the distance that would result in the maximum ground-level 95th percentile X/Q values for the elevated release. Given the FitzPatrick plant layout, topography, and meteorological data, this distance is approximately 405 meters. This approach is reasonable, and the staff finds it acceptable. The resultant X/Q values used by the licensee and the staff are given in Table 3 (Attachment 3).

2.2 Conclusion on the Technical Evaluation

The staff has reviewed the licensee's analysis and performed a confirmatory calculation of the radiological consequence resulting from the postulated LOCA. The doses calculated by the staff and the licensee are listed in Table 2 (Attachment 2). As shown in the table, the doses calculated by the licensee are comparable to those calculated by the staff, and they are well within the relevant acceptable dose criteria. Considering the many uncertainties in the modeling of fission product transport and removal mechanisms, the staff concludes that the differences in the doses calculated by the staff and the licensee are not significant. Therefore, the staff concludes that the radiological consequences analyzed and submitted by the licensee are acceptable.

The staff also finds that the decrease of the primary coolant sampling requirement threshold value to 0.002 µCi/ml from 0.007 µCi/ml of dose-equivalent iodine-131 requested by the licensee is acceptable because this change is more conservative for monitoring iodine concentration in the primary coolant and for assessing the radiological consequences.

On the basis of this evaluation, the staff concludes that the license amendment requested by the licensee is acceptable. However, the acceptance of this amendment does not relieve the licensee from the responding to the U.S. Nuclear Regulatory Commission (NRC) Generic Letter 99-02 or regulatory actions that may be proposed in the future as the NRC-Nuclear Energy Institute Task Force resolves the control room habitability generic issues.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York 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 19977). The amendment also relates to changes in recordkeeping, reporting, or administrative procedures or requirements. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (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.

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.

Attachments:
1. Table 1 Parameters and Assumptions Used in Radiological Consequence Calculations
2. Table 2 Radiological Consequences (Thyroid doses in rem)
3. Meteorological Data

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TABLE 1

Parameters and Assumptions Used in Radiological Consequence Calculations

Parameter	Value
Reactor power	2587 Mwt
Source term	Regulatory Guide 1.3
Dose conversion factors	ICRP-30
Computer code used	HABIT, Version 1.1
Drywell volume	1.5E+5 ft^3
Containment/MSIV leak rate	1.5 percent per day
Standby Gas Treatment System	
Iodine removal efficiency	90 percent
Reactor building mixing/holdup	0
ECCS Leak Rate	5.0 gpm
Iodine partition factor	10
Control room	
Volume	1.49E+5 ft^3
Unfiltered inleakage	
Before isolation	1.5E+4 ft^3/min
After isolation	2.11E+3 ft^3/min
Isolation time	30 minutes
Make-up air flow rate	1.0E+3 ft^3/min
Iodine removal efficiency	90 percent

TABLE 2

Radiological Consequences
(Thyroid Doses in rem)

Pathways	EAB[1]		LPZ[2]		Control Room	
	NRC	FitzPatrick	NRC	FitzPatrick	NRC	FitzPatrick
Containment/MSIV Leak	62.4	58.2	67.7	63.2	8.0	10.1
ECCS Leak	6.5	3.99	7.0	5.50	0.14	1.06
TOTAL	68.9	62.2	74.7	68.7	8.14	11.2
Dose criteria	300[3]		300[3]		30[4]	

Radiological Consequences
(Whole Body Doses in rem)

Pathways	EAB[1]		LPZ[2]		Control Room	
	NRC	FitzPatrick	NRC	FitzPatrick	NRC	FitzPatrick
Containment/MSIV Leak	1.6	2.32	1.1	1.86	< 1	< 1
ECCS Leak	< 1	< 1	< 1	< 1	< 1	< 1
TOTAL	1.6	2.32	1.1	1.86	< 1	< 1
Dose criteria	25[3]		25[3]		5[4]	

(1) Exclusion Area Boundary
(2) Low Population Zone
(3) 10 CFR Part 100
(4) General Design Criteria 19

TABLE 3

Meteorological Data

Exclusion Area Boundary

Time (hr)	X/Q (sec/m^3)
0-2	5.24×10^{-5}

Low Population Zone

Time (hr)	X/Q (sec/m^3)
0-4	2.04×10^{-5}
4-8	2.17×10^{-6}
8-24	9.53×10^{-7}
24-96	3.90×10^{-7}
96-720	1.08×10^{-7}

Control Room

Time (hr)	X/Q (sec/m^3)
0-8	9.26×10^{-7}
8-24	6.75×10^{-7}
24-96	3.39×10^{-7}
96-720	1.26×10^{-7}