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

October 15, 1993

Docket No. 50-286

Mr. Ralph E. Beedle
Executive Vice President - Nuclear Generation
Power Authority of the State of New York
123 Main Street
White Plains, New York 10601

Dear Mr. Beedle:

SUBJECT: ISSUANCE OF AMENDMENT FOR INDIAN POINT NUCLEAR GENERATING UNIT NO. 3 (TAC NO. M86380)

The Commission has issued the enclosed Amendment No.139 to Facility Operating License No. DPR-64 for the Indian Point Nuclear Generating Unit No. 3. The amendment consists of changes to the Technical Specifications (TS) in response to your application transmitted by letter dated May 4, 1993, as supplemented September 29, 1993.

The amendment revises Technical Specifications (TS) Sections 3.2 (Chemical and Volume Control System), 3.3 (Engineered Safety Features), and 4.1 (Operational Safety Review) to eliminate the requirement to maintain a Boron Injection Tank (BIT) and the related heat tracing. In addition, the amendment revises TS Section 4.4 (Containment Tests) and the associated Bases to remove the containment temperature reference in the containment leak test acceptance criteria.

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,

Nicola F. Conicella, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 139to DPR-64
2. Safety Evaluation

cc w/enclosures:
See next page

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Mr. Ralph E. Beedle
Power Authority of the State
of New York

Indian Point Nuclear Generating
Station Unit No. 3

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DATED: October 15, 1993

AMENDMENT NO. 139 TO FACILITY OPERATING LICENSE NO. DPR-64-INDIAN POINT UNIT 3

Docket File
NRC & Local PDRs
PDI-1 Reading
S. Varga, 14/E/4
J. Calvo, 14/A/4
R. Capra
C. Vogan
N. Conicella
OGC
D. Hagan, 3302 MNBB
G. Hill (2), P1-22
C. Grimes, 11/F/23
ACRS (10)
OPA
OC/LFDCB
PD plant-specific file
C. Cowgill, Region I
R. Jones, 8/E/21
C. Jackson, 8/E/23

cc: Plant Service list

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

POWER AUTHORITY OF THE STATE OF NEW YORK

DOCKET NO. 50-286

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 139
License No. DPR-64

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

   A. The application for amendment by Power Authority of the State of New York (the licensee) dated May 4, 1993, as supplemented September 29, 1993, 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.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-64 is hereby amended to read as follows:

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(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 139, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance to be implemented prior to restart from the current outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Capra

Robert A. Capra, Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 15, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 139

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

Revise Appendix A as follows:

Remove Pages	Insert Pages
3.2-1	3.2-1
3.2-2	3.2-2
3.2-3	3.2-3
3.2-4	3.2-4
3.3-2	3.3-2
3.3-4	3.3-4
3.3-15	3.3-15
4.4-2	4.4-2
4.4-7	4.4-7
4.4-10	4.4-10
Table 4.1-1 (sheet 3 of 6)	Table 4.1-1 (sheet 3 of 6)
Table 4.1-2 (sheet 1 of 2)	Table 4.1-2 (sheet 1 of 2)

3.2 CHEMICAL AND VOLUME CONTROL SYSTEM

Applicability

Applies to the operational status of the Chemical and Volume Control System.

Objective

To define those conditions of the Chemical and Volume Control System necessary to ensure safe reactor operation.

Specification

A. When fuel is in the reactor there shall be at least one flow path to the core for boric acid injection.

B. The reactor shall not be brought above the cold shutdown condition unless the following requirements are met:

1. Two charging pumps shall be operable.

2. Two boric acid transfer pumps shall be operable.

3. The boric acid storage system shall contain a minimum of 6100 gallons of 11 1/2% to 13% by weight (20,112 ppm to 22,735 ppm of boron) boric acid solution at a temperature of at least 145°F.

4. System piping and valves shall be operable to the extent of establishing one flow path from the boric acid storage system and one flow path from the refueling water storage tank (RWST) to the Reactor Coolant System.

5. The appropriate boric acid storage tank level indicator(s) shall be operating.

6. Two channels of heat-tracing shall be operable for the flow path from the boric acid storage system to the Reactor Coolant System.

Amendment No. ~~78~~, ~~88~~, ~~119~~, 139

7. City water piping and valves shall be operable to the extent required to provide emergency cooling water to the charging pumps and flush water for the concentrated boric acid piping from the outlet of the boric acid storage tanks to the charging pump suction.

C. The requirements of 3.2.B may be modified to allow any one of the following components to be inoperable at any one time:

1. One of the two operable charging pumps may be removed from service provided a second charging pump is restored to an operable status within 24 hours.

2. One boric acid transfer pump may be inoperable for a period not to exceed 48 hours.

3. The boric acid storage system may be inoperable for a period not to exceed 48 hours provided that the RWST is operable.

4. One channel of heat tracing for the flow path from the boric acid storage system to the Reactor Coolant System may be out of service provided the failed channel is restored to an operable status within 7 days and the redundant channel is demonstrated to be operable daily during that period.

D. If the Chemical and Volume Control System is not restored to meet the requirements of 3.2.B within the time period specified in 3.2.C, then:

1. If the reactor is critical, it shall be brought to the hot shutdown condition utilizing normal operating procedures. The shutdown shall start no later than at the end of the specified time period.

2. If the reactor is subcritical, the reactor coolant system temperature and pressure shall not be increased more than 25°F and 100 psi, respectively, over existing values.

3. In either case, if the requirements of 3.2.B are not satisfied within an additional 48 hours, the reactor shall be brought to the cold shutdown condition utilizing normal operating procedures. The shutdown shall start no later than the end of the 48 hour period.

BASIS

The Chemical and Volume Control System[1] provides control of the Reactor Coolant System boron inventory. This is normally accomplished by using any one of the three charging pumps in series with either one of the two boric acid transfer pumps. An alternate method of boration will be to use the charging pumps taking suction directly from the refueling water storage tank. A third method will be to depressurize and use the safety injection pumps.

There are three sources of borated water available for injection through 3 different paths:

1. The boric acid transfer pumps can deliver the contents of the boric acid storage system to the charging pumps.

2. The charging pumps can take suction from the refueling water storage tank.

3. Injection of borated water from the refueling water storage tank with the safety injection pumps[2].

The quantity of boric acid in storage from either the boric acid storage system or the refueling water storage tank is sufficient to borate the reactor coolant in order to reach cold shutdown at any time during core life.

A combined minimum deliverable volume of 6100 gallons with an averaged concentration of the 11 1/2% to 13% by weight (20,112 ppm to 22,735 ppm of boron) of boric acid are required to meet cold shutdown conditions. An upper concentration limit of 13% (22,735 ppm of boron) boric acid in the boric acid storage system is specified to maintain solution solubility at the specified low temperature limit of 145°F. One channel of heat tracing is sufficient to maintain the specified low temperature limit. The second channel of heat tracing provides backup for continuous plant operation when one channel is inoperable. Should both channels of heat tracing become inoperable, the reactor will be shutdown and can easily be borated before the line temperature is reduced near the boric acid precipitive temperature.

The city water system is used as a source of water for emergency cooling of the charging pumps and as a source of flush water to remove concentrated boric acid from the piping between the outlet of the boric acid storage tanks and the inlet to the charging pumps in the unlikely event of a complete loss of electrical power and/or a complete loss of service water resulting from turbine missiles.

<u>References</u>

1) FSAR - Section 9.2
2) FSAR - Section 6.2
3) "Revised Feasibility Report For BIT Elimination For Indian Point Unit 3," July 1988 (Westinghouse report).

c. One residual heat removal pump and heat exchanger together with the associated piping and valves operable.

d. One recirculation pump together with its associated piping and valves operable.

2. If the Safety Injection and Residual Heat Removal Systems are not restored to meet the requirements of 3.3.A.1 within 1 hour the reactor shall be in the cold shutdown condition within the next 20 hours.

3. The reactor coolant system T_{avg} shall not exceed 350°F unless the following requirements are met:

a. The refueling water storage tank contains a minimum of 346,870 gallons of water at a boron concentration ≥2400 ppm and ≤2600 ppm.

b. DELETED

c. The four accumulators are pressurized between 600 and 700 psig and each contains a minimum of 775 ft^3 and a maximum of 815 ft^3 of water at a boron concentration ≥2000 ppm and ≤2600 ppm. Accumulator isolation valves 894A, B, C, and D shall be open and their power supplies deenergized whenever the reactor coolant system pressure is above 1000 psig.

a. The accumulators may be isolated during the performance of the reactor coolant system hydrostatic tests.

   For the purpose of accumulator check valve leakage testing, one accumulator may be isolated at a time, for up to 8 hours, provided the reactor is in the hot shutdown condition.

b. One safety injection pump may be out of service, provided the pump is restored to an operable status within 24 hours.

c. One residual heat removal pump may be out of service, provided the pump is restored to an operable status within 24 hours.

d. One residual heat exchanger may be out of service provided that it is restored to an operable status within 48 hours.

e. Any valve required for the functioning of the system during and following accident conditions may be inoperable provided that it is restored to an operable status within 24 hours and all valves in the system that provide the duplicate function are operable.

f. DELETED

g. One refueling water storage tank low level alarm may be inoperable for up to 7 days provided the other low level alarm is operable.

cold shutdown condition, utilizing normal shutdown and cooldown procedures. In the cold shutdown condition there is no possibility of an accident that would release fission products or damage the fuel elements.

The plant operating procedures require immediate action to effect repairs of an inoperable component, and, therefore, in most cases repairs will be completed in less than the specified allowable repair times. The limiting times to repair are based on two considerations:

1) Assuring with high reliability that the safeguard system will function properly if required to do so.

2) Allowances of sufficient time to effect repairs using safe and proper procedures.

Assuming the reactor has been operating at full rated power, the magnitude of the decay heat decreases after initiating hot shutdown. Thus, the requirement for core cooling in case of a postulated loss-of-coolant accident while in the hot shutdown condition is significantly reduced below the requirements for a postulated loss-of-coolant accident during power operation. Putting the reactor in the hot shutdown condition significantly reduces the potential consequences of a loss-of-coolant accident, and also allows more free access to some of the engineered safeguards components in order to effect repairs.

Failure to complete repairs within 1 hour of going to the hot shutdown condition is considered indicative of a requirement for major maintenance and, therefore, in such a case the reactor is to be put into the cold shutdown condition.

The limits for the Refueling Water Storage Tank and the accumulators insure the required amount of water with the proper boron concentration for injection into the reactor coolant system following a loss-of-coolant accident is available. These limits are based on values used in the accident analysis.[9][13]

d. Closure of the containment isolation valves for the purpose of the test shall be accomplished by the means provided for normal operation of the valves.

2. Acceptance Criteria

The measured leakage rate shall be less than 0.75 L_a where L_a is equal to 0.1 w/o per day of containment steam air atmosphere at 42.42 psig.

3. Frequency

A set of three leakage rate tests shall be performed (during plant shutdown), at approximately equal intervals during each 10-year service period. The third test of each set shall be conducted when the plant is shutdown for the 10-year plant in service inspection.

B. DELETED

Basis

The containment is designed for a pressure of 47 psig. [1] While the reactor is operating, the internal environment of the containment will be air at essentially atmospheric pressure and an average maximum temperature of approximately 130°F. The limiting peak containment temperature, based on LOCA containment response, is 261.5°F. [7] The peak containment pressure, also based on LOCA containment response, is 42.29 psig. [7] The acceptance criteria of specification 4.4.A.2. was changed by amendment 98 to reflect analysis [4] done for the ultimate heat sink temperature increase. As stated, the current peak pressure, calculated for high head safety injection flow balancing, is 42.29 psig. The acceptance criteria of 42.42 psig is conservative with respect to the current calculated peak pressure of 42.29.

Prior to initial operation, the containment was strength-tested at 54 psig and was leak-tested. The acceptance criterion for this pre-operational leakage rate test was established as 0.075 W/o (.75 L_a) per 24 hours at 40.6 psig and 263°F, which were the peak accident pressure and temperature conditions at that time. This leakage rate is consistent with the construction of the containment, [2] which is equipped with a Weld Channel and Penetration Pressurization System for continuously pressurizing both the penetrations and the channels over all containment liner welds. These channels were independently leak-tested during construction.

The safety analysis has been performed on the basis of a leakage rate of 0.10 W/o per day for 24 hours. With this leakage rate and with minimum containment engineered safeguards operating, the public exposure would be well below 10CFR100 values in the event of the design basis accident.[3]

The performance of a periodic integrated leakage rate test during plant life provides a current assessment of potential leakage from the containment in case of an accident that would pressurize the interior of the containment. In order to provide a realistic appraisal of the integrity of the containment under accident conditions, the containment isolation valves are to be closed in the normal manner and without preliminary exercising or adjustments.

These specifications have been developed using Appendix J (issue effective date March 16, 1973) of 10CFR50 (with the surveillance frequency exception noted previously) and ANSI N45.4-1972 "Leakage Rate Testing of Containment structures for Nuclear Reactors" (March 16, 1972) for guidance.

The maximum permissible inleakage rate from the containment isolation valves sealed with service water for the full 12-month period of post accident recirculation without flooding the internal recirculation pumps is 0.36 gpm per fan cooler.

REFERENCES

(1) FSAR - Section 5
(2) FSAR - Section 5.1.7
(3) FSAR - 14.3.5
(4) WCAP - 12269 Rev. 1, "Containment Margin Improvement Analysis for IP-3 Unit 3"
(5) FSAR - Section 6.6
(6) FSAR - Section 6.5
(7) SECL-92-131, Indian Point Unit 3 High Head Safety Injection Flow Changes Safety Evaluation, June 1992

TABLE 4.1-1 (Sheet 3 of 6)

Channel Description	Check	Calibrate	Test	Remarks
16. Containment Water Level Monitoring System:				
a. Containment Sump	N.A.	18M	N.A.	Narrow Range, Analog
b. Recirculation Sump	N.A.	18M	N.A.	Narrow Range, Analog
c. Containment Water Level	N.A.	18M	N.A.	Wide Range
17. Accumulator Level and Pressure	S***	18M	N.A.	
18. Steam Line Pressure	S	18M	Q	
19. Turbine First Stage Pressure	S	18M	Q	
20. Reactor Protection Relay Logic	N.A.	N.A.	TM	
21. Turbine Trip Low Auto Stop Oil Pressure	N.A.	18M	N.A.	
22. DELETED	DELETED	DELETED	DELETED	
23. Temperature Sensor in Auxiliary Boiler Feedwater Pump Building	N.A.	N.A.	18M	
24. Temperature Sensors in Primary Auxiliary Building				
a. Piping Penetration Area	N.A.	N.A.	24M	
b. Mini-Containment Area	N.A.	N.A.	24M	
c. Steam Generator Blowdown Heat Exchanger Room	N.A.	N.A.	24M	

TABLE 4.1-2 (Sheet 1 of 2)

FREQUENCIES FOR SAMPLING TESTS			
Sample	Analysis	Frequency	Maximum Time Between Analysis
1. Reactor Coolant	Gross Activity[1]	5 days/week[1][4]	3 days[4]
	Tritium Activity	Weekly[1]	10 days
	Boron concentration	2 days/week	5 days
	Radiochemical (gamma)[2] Spectral Check	Monthly	45 days
	Oxygen and Chlorides Concentration	3 times per 7 days	3 days
	Fluorides Concentration	Weekly	10 days
	$\bar{E}$ Determination [3]	Semi-Annually	30 weeks
	Isotopic Analysis for I-131, I-133, I-135	Once per 14 days[5]	20 days
2. Boric Acid Tank	Boron Concentration, Chlorides	Weekly	10 days
3. Spray Additive Tank	NaOH Concentration	Monthly	45 days
4. Accumulators	Boron Concentration	Monthly	45 days
5. Refueling Water Storage Tank	Boron Concentration pH, Chlorides	Monthly	45 days
	Gross Activity	Quarterly	16 weeks
6. Secondary Coolant	I-131 Equivalent (Isotopic Analysis)	Monthly	45 days
	Gross Activity	3 times per 7 days	3 days
7. Component Cooling Water	Gross Activity, Corrosion Inhibitor and pH	Monthly	45 days
8. Spent Fuel Pool (when fuel stored)	Gross Activity Boron Concentration, Chlorides	Monthly	45 days

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 139 TO FACILITY OPERATING LICENSE NO. DPR-64

POWER AUTHORITY OF THE STATE OF NEW YORK

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

DOCKET NO. 50-286

1.0 INTRODUCTION

By letter dated May 4, 1993, as supplemented September 29, 1993, the Power Authority of the State of New York (the licensee) submitted a request for changes to the Indian Point Nuclear Generating Unit No. 3 (IP3), Technical Specifications (TS). The requested changes would revise TS Sections 3.2 (Chemical and Volume Control System), 3.3 (Engineered Safety Features), and 4.1 (Operational Safety Review) to eliminate the requirement to maintain a Boron Injection Tank (BIT) and the related heat tracing. In addition, the requested changes would also revise TS Section 4.4 (Containment Tests) and the associated Bases to remove the containment temperature reference in the containment leak test acceptance criteria. The September 29, 1993, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

2.1 Boron Injection Tank

The BIT provides a highly concentrated source of borated water for negative reactivity insertion into the core. The BIT is located between the high head charging pumps and the core as part of the Safety Injection System. Injection through the BIT is intended to mitigate the effects of a rapid decrease in reactor coolant system (RCS) temperature following rupture of a main steam line. Due to the negative moderator temperature coefficient, the cooldown causes an increase in the core reactivity. This increase in core reactivity, coupled with the possibility that the most reactive control rod is stuck in the fully withdrawn position, makes departure from nucleate boiling (DNB) in the core possible. The BIT is designed to supply the RCS with enough negative reactivity to prevent the reactor from exceeding the departure from nucleate boiling ratio (DNBR) limits during rapid cooldowns.

Analysis of rapid RCS cooldown by the licensee, using improved methods approved by the staff, has shown that the negative reactivity supplied by the BIT is not necessary to avoid DNB. The analysis that the licensee conducted took into account five different scenarios. The five scenarios covered are:

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1. Steam pipe severance, upstream of the flow restrictor, with offsite power available.

2. Steam pipe severance, upstream of the flow restrictor, without offsite power available.

3. Steam pipe severance, downstream of the flow restrictor, with offsite power available.

4. Steam pipe severance, downstream of the flow restrictor, without offsite power available.

5. A failed secondary safety or relief valve in the open position, with offsite power available.

The transient analysis was performed with the LOFTRAN computer code. The analysis assumed the most reactive cluster control assembly remained in the fully withdrawn position following the reactor trip. The plant was assumed to be initially at hot zero power with the minimum required shutdown margin. For purposes of conducting a conservative, bounding analysis, the licensee assumed a 24% uniform steam generator tube plugging level to determine the core flow at the time of the break, and a 0% tube plugging level in the determination of the heat transfer surface area. The higher assumed heat transfer surface area causes a more rapid cooldown of the primary system and, therefore, a more severe transient. The four steam generators at IP3 were replaced during the 1989 refueling outage and the current configuration has approximately 1.4 - 1.5% "equivalent tube plugging" in each steam generator. The reduced core flow and increased surface area associated with 24% and 0% tube plugging contradict each other; however, each has been applied in the analysis in a conservative fashion so as to bound actual steam generator conditions.

The results of the licensee's analysis show that the reactor becomes critical and returns to power because of the assumed large end of life (EOL) moderator temperature coefficient. Power increases until borated water from the refueling water storage tank (RWST) is supplied to the core through the safety injection system. In all five of the scenarios the DNBR limit is met (i.e., not exceeded). There are no fuel failures predicted. For the steam pipe severance cases, scenarios 1-4, which are considered hypothetical breaks, the calculated radiation releases are within the 10 CFR Part 100 limits. For the failed secondary safety or relief valve, scenario 5, which is considered a credible break, the radiation releases are within the 10 CFR Part 20 limits.

The NRC staff has reviewed the information presented by the licensee regarding the removal of the BIT and the associated heat tracing and concludes that the requested changes are acceptable. Analysis performed shows that the DNB limits are not exceeded and that the radiation releases from the plant will remain within the applicable limits. The staff has reviewed the licensee's analysis and has found that the analytical results are reasonable and

appropriate. In addition, the proposed BIT removal is consistent with the staff's position as stated in Generic Letter 85-16, "High Boron Concentration," which addressed BIT removal.

2.2 Containment Leak Test

TS 4.4.A.2 contains the acceptance criteria for the Containment integrated leakage rate test. The current TS acceptance criteria is "... less than 0.75 L_a where L_a is equal to 0.1 w/o per day of containment steam air atmosphere at 42.42 psig and 257 °F, which are the peak accident pressure and temperature conditions." The licensee proposed removing the peak containment temperature reference from the acceptance criteria and placing it in the associated TS Bases. The licensee stated that this containment leak test was never intended to be performed at the temperature indicated in the acceptance criteria. The temperature only indicates the postulated peak accident temperature determined by the containment response analysis. The NRC staff has reviewed the information presented by the licensee regarding the containment integrated leakage rate testing acceptance criteria and concludes that the proposed change is acceptable. In addition, the licensee's proposed change is consistent with NUREG 1431, "Standard Technical Specifications for Westinghouse Plants," which clearly indicates that the containment leak test was intended to be performed at the calculated peak containment pressure (P_a) not at the peak containment temperature.

As stated above, the peak containment temperature information would be moved to the associated TS Bases. However, the TS Bases would reflect an updated peak containment temperature of 261.5 °F. In addition, the TS Bases would indicate an updated peak containment pressure of 42.29 psig. These updated values are a result of analysis SECL-92-131, "Indian Point Unit 3 High Head Safety Injection Flow Changes Safety Evaluation," dated June 1992, which was done to support the safety injection flow balance test conducted during the 1992 (cycle 8/9) refueling outage. Although the analysis indicated a peak pressure of 42.29 psig, the acceptance criteria of TS 4.4.A.2 will now be more conservative since it will remain unchanged at 42.42 psig. The NRC staff has reviewed the information presented by the licensee regarding the containment integrated leakage rate test Bases and offers no objections.

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 (58 FR 34089). 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:
Christopher Jackson

Date: October 15, 1993