March 19, 1990

Docket No. 50-286

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Mr. John C. Brons
Executive Vice President - Nuclear Generation
Power Authority of the State of New York
123 Main Street
White Plains, New York 10601

Dear Mr. Brons:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. 72907)

The Commission has issued the enclosed Amendment No. 93 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 in response to your application transmitted by letter dated March 10, 1989, as supplemented by letter dated February 9, 1990.

The amendment revises the Technical Specifications to (1) allow less frequent testing of turbine steam stop and control valves and (2) eliminate Limiting Conditions for Operation and Surveillance Requirements for the turbine independent electrical overspeed protection system.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular bi-weekly Federal Register notice.

Sincerely,

Joseph D. Neighbors, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 93 to DPR-64
2. Safety Evaluation

cc: w/enclosures
See next page

OFC	PDI-1	PDI-1	EMTB	SPLB	OGC	PDI-1
NAME	CVogan	JNeighbors:rc	CYCheng	CMcCracken		RACapra
DATE	2/13/90	2/14/90	2/16/90	2/27/90	2/28/90	3/19/90

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Mr. John C. Brons
Power Authority of the State
of New York

Indian Point 3 Nuclear Power Plant

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

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. 93
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 March 10, 1989, as supplemented by letter dated February 9, 1990, 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. 93 , 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.

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: March 19,1990

ATTACHMENT TO LICENSE AMENDMENT NO. 93

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

Revise Appendix A as follows:

Remove Pages	Insert Pages
Table 3.5-2 (sheet 1 of 3)	Table 3.5-2 (sheet 1 of 2)
Table 3.5-2 (sheet 2 of 3)	Table 3.5-2 (sheet 2 of 2)
Table 3.5-2 (sheet 3 of 3)	
Table 4.1-1 (sheet 1 of 5)	Table 4.1-1 (sheet 1 of 5)
Table 4.1-1 (sheet 2 of 5)	Table 4.1-1 (sheet 2 of 5)
Table 4.1-1 (sheet 3 of 5)	Table 4.1-1 (sheet 3 of 5)
Table 4.1-1 (sheet 4 of 5)	Table 4.1-1 (sheet 4 of 5)
Table 4.1-1 (Sheet 5 of 5)	Table 4.1-1 (Sheet 5 of 5)
Page 4.1-3	Page 4.1-3
Table 4.1-3 (sheet 1 of 2)	Table 4.1-3 (sheet 1 of 2)
Table 4.1-3 (sheet 2 of 2)	Table 4.1-3 (sheet 2 of 2)

TABLE 3.5-2 (Sheet 1 of 3)

REACTOR TRIP INSTRUMENTATION LIMITING OPERATING CONDITIONS					
NO. FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MIN. NUMBER OF OPERABLE CHANNELS	4 MIN. DEGREE OF REDUNDANCY	5 OPERATOR ACTION IF CONDITIONS OF COL. 3 OR 4 CANNOT BE MET*
1. Manual Reactor Yrip	2	1	1	0	Maintain hot shutdown
2. Nuclear Flux Power Range	4	2	3	2	Maintain hot shutdown
	4	2	2	1	For zero power physics tests only
3. Overtemperature ΔT	4	2	3	2	Maintain hot shutdown
4. Overpower ΔT	4	2	3	2	Maintain hot shutdown
5. Low Pressurizer Pressure	4	2	3	2	Maintain hot shutdown
6. Hi Pressurizer Pressure	3	2	2	1	Maintain hot shutdown
7. Pressurizer-Hi Water Level	3	2	2	1	Maintain hot shutdown
8. Low Flow One Loop (Power ≥ P-8)	3/loop	2/loop (any loop)	2/operable loop	1/operable loop	Maintain hot shutdown
Low Flow Two Loops (Power < P-8 and ≥ P-10)	3/loop	2/loop (any two loops)	2/operable loop	1/operable loop	Maintain hot shutdown

TABLE 3.5-2 (Sheet 2 of 3)

REACTOR TRIP INSTRUMENTATION LIMITING OPERATING CONDITIONS					
NO. FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MINIMUM OPERABLE CHANNELS	4 MINIMUM DEGREE OF REDUNDANCY	5 OPERATOR ACTION IF CONDITIONS OF COLUMNS 3 OR 4 CANNOT BE MET*
9. Lo Lo Steam Generator Water Level	3/loop	2/loop	2/loop	1/loop	Maintain hot shutdown
10. Undervoltage 6.9 KV Bus	1/bus	2	3	2	Maintain hot shutdown
11. Low Frequency 6.9 KV Bus**	1/bus	2	3	2	Maintain hot shutdown
12. Turbine Trip: Low auto stop oil pressure	3	2	2	1	Maintain reactor power below 10% of full power
13. Reactor Trip Breakers***	2	1	2	1	Maintain hot shutdown****
14. Reactor Protection Relay Logic	2	1	2	1	Maintain hot shutdown****

* Maintain hot shutdown means maintain or proceed to hot shutdown within 4 hours using normal operating procedures, if the unacceptable condition arises during operation.

** 2/4 trips all four reactor coolant pumps.

*** A reactor trip breaker is considered inoperable if any of its components fail to meet test specifications. If either the undervoltage or shunt trip device (not both) prevent a breaker from proper operation, then 72 hours are allowed to restore the failed device to operable status before the affected breaker is declared inoperable.

TABLE 3.5-2 (Sheet 3 of 3)

**** Upon proceeding to hot shutdown as a result of an inoperable reactor trip breaker or relay logic, 48 hours are allowed to restore the minimum number of operable channels required by column 3. If minimum operability is not restored after this 48 hour period, rod withdrawal capability shall be defeated within one hour by opening one reactor trip breaker and its associated bypass breaker or isolating power to the Control Rod Drive System.

TABLE 4.1-1 (Sheet 1 of 5)

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTS OF INSTRUMENT CHANNELS

Channel Description	Check	Calibrate	Test	Remarks
1. Nuclear Power Range	S	D (1) M (3)*	M (2)** M (4)	1) Heat balance calibration 2) Bistable action (permissive, rod stop, trips) 3) Upper and lower chambers for axial offset 4) Signal to ΔT
2. Nuclear Intermediate Range	S (1)	N.A.	P (2)	1) Once/shift when in service 2) Verification of channel response to simulated inputs
3. Nuclear Source Range	S (1)	N.A.	P (2)	1) Once/shift when in service 2) Verification of channel response to simulated inputs
4. Reactor Coolant Temperature	S	R	M (1) M (2)	1) Overtemperature - ΔT 2) Overpower - ΔT
5. Reactor Coolant Flow	S	R	M	
6. Pressurizer Water Level	S	R	M	
7. Pressurizer Pressure	S	R	M	High and Low
8. 6.9 KV Voltage & Frequency	N.A.	R	M	Reactor protection circuits only
9. Analog Rod Position	S	R	M	

TABLE 4.1-1 (Sheet 2 of 5)

Channel Description	Check	Calibrate	Test	Remarks
10. Steam Generator Level	S	R	M	
11. Residual Heat Removal Pump Flow	N.A.	R	N.A.	
12. Boric Acid Tank Level	S	R	N.A.	Bubbler tube rodded during calibration
13. Refueling Water Storage Tank Level	W	R	N.A.	Low level alarms
14. Containment Pressure	S	R	M	High and High-High
15. Process and Area Radiation Monitoring Systems	D	R	Q	
16. Containment Water Level Monitoring System:				
a. Containment Sump	N.A.	R	N.A.	Narrow Range, Analog
b. Recirculation Sump	N.A.	R	N.A.	Narrow Range, Analog
c. Containment Water Level	N.A.	R	N.A.	Wide Range
17. Accumulator Level and Pressure	S***	R	N.A.	
18. Steam Line Pressure	S	R	M	
19. Turbine First Stage Pressure	S	R	M	
20. Reactor Protection Relay Logic	N.A.	N.A.	TM	
21. Turbine Trip Low Auto Stop Oil Pressure	N.A.	R	N.A.	
22. Boron Injection Tank Return Flow	S	R	N.A.	

TABLE 4.1-1 (Sheet 3 of 5)

Channel Description	Check	Calibrate	Test	Remarks
23. Temperature Sensor in Auxiliary Boiler Feedwater Pump Building	N.A.	N.A.	R	
24. Temperature Sensors in Penetration Area of Primary Auxiliary Building	N.A.	N.A.	R	
25. Level Sensors in Turbine Building	N.A.	N.A.	R	
26. Volume Control Tank Level	N.A.	R	N.A.	
27. Boric Acid Makeup Flow Channel	N.A.	R	N.A.	
28. Auxiliary Feedwater:				
a. Steam Generator Level	S	R	M	Low-Low
b. Undervoltage	N.A	R	R	
c. Main Feedwater Pump Trip	N.A	N.A.	R	
29. Reactor Coolant System Subcooling Margin Monitor	D	R	N.A.	
30. PORV Position Indicator	N.A.	R	R	Limit Switch
31. PORV Position Indicator	D	R	R	Acoustic Monitor
32. Safety Valve Position Indicator	D	R	R	Acoustic Monitor
33. Auxiliary Feedwater Flow Rate	N.A.	R.	N.A.	

TABLE 4.1-1 (Sheet 4 of 5)

Channel Description	Check	Calibrate	Test	Remarks
34. Plant Effluent Radioiodine/ Particulate Sampling	N.A.	N.A.	R	Sample line common with monitor R-13
35. Loss of Power				
a. 480v Bus Undervoltage Relay	N.A.	R	M	
b. 480v Bus Degraded Voltage Relay	N.A.	R	M	
c. 480v Safeguards Bus Undervoltage Alarm	N.A.	R	M	
36. Main Steam Line Radiation Monitors	D	R	Q	R-62A, B,C,D
37. Containment Hydrogen Monitors	D	Q	M	
38. Wide Range Plant Vent Monitor	D	R	Q	R-27
39. High Range Containment Radiation Monitors	D	R	Q	R-25, R-26
40. Core Exit Thermocouples	D	N.A.	N.A.	
41. Overpressure Protection System (OPS)	D	R	R	
42. Reactor Trip Breakers	N.A.	N.A.	TM(1)	1) Independent operation of undervoltage and shunt trip attachments
			R(2)	2) Independent operation of undervoltage and shunt trip from Control Room manual push-button

TABLE 4.1-1 (Sheet 5 of 5)

Channel Description	Check	Calibrate	Test	Remarks
43. Reactor Trip Bypass Breakers	N.A.	N.A.	(1)	1) Manual shunt trip prior to each use
			R(2)	2) Independent operation of undervoltage and shunt trip from Control Room manual push-button
			R(3)	3) Automatic undervoltage trip
44. Reactor Vessel Level Indication System (RVLIS)	D	R	N.A.	

* By means of the movable incore detector system

** Monthly when reactor power is below the setpoint and prior to each startup if not done previous month.

*** If either an accumulator level or pressure instrument channel is declared inoperable, the remaining level or pressure channel must be verified operable by interconnecting and equalizing (pressure and/or level wise) a minimum of two accumulators and crosschecking the instrumentation.

S - Each shift
P - Prior to each startup if not done previous week
NA- Not applicable
D - Daily
TM- At least every two months on a staggered test basis (i.e., one train per month)
W - Weekly
M - Monthly
Q - Quarterly
R - Each refueling outage

2.5×10^{-6} failure/hrs. per channel. This is based on operating experience at conventional and nuclear plants. An unsafe failure is defined as one which negates channel operability and which, due to its nature, is revealed only when the channel is tested or attempts to respond to a bona fide signal.

For a specified test interval W and an M out of N redundant system with identical and independent channels having a constant failure rate λ, the average availability A is given by:

$$A = \frac{W - Q\left(\frac{W}{N-M+2}\right)}{W} = 1 - \frac{N!}{(N-M+2)!\,(N-1)!}(\lambda W)^{N-M+1}$$

where A is defined as the fraction of time during which the system is functional, and Q is the probability of failure of such a system during a time interval W.

For a 2-out-of-3 system A = 0.9999968, assuming a channel failure rate, λ, equal to 2.5×10^{-6} hr^{-1} and a test interval, W, equal to 720 hrs.

This average availability of the 2-out-of-3 system is high, hence the test interval of one month is acceptable.

Because of their greater degree of redundancy, the 1/3 and 2/4 logic arrays provide an even greater measure of protection and are thereby acceptable for the same testing interval. Those items specified for monthly testing are associated with process components where other means of verification provide additional assurance that the channel is operable, thereby requiring less frequent testing.

The Turbine Steam Stop and Control Valves shall be tested at a frequency determined by the methodology presented in WCAP-11525, "Probabilistic Evaluation of Reduction in Turbine Valve Test Frequency", and in accordance with established NRC acceptance criteria for the probability of a missile ejection incident at IP-3. In no case shall the test interval for these valves exceed one year.

TABLE 4.1-3 (Sheet 1 of 2)

FREQUENCIES FOR EQUIPMENT TESTS			
		Check	Frequency
1.	Control Rods	Rod drop times of all control rods	R
2.	Control Rods	Partial movement of all control rods	Every 2 weeks during reactor critical operations
3.	Pressurizer Safety Valves	Set Point	R
4.	Main Steam Safety Valves	Set Point	R
5.	Containment Isolation System	Automatic actuation	R
6.	Refueling System Interlocks	Functioning	R (Prior to movement of core components)
7.	Primary System Leakage	Evaluate	5 days/week
8.	Diesel Generators Nos. 31, 32, & 33 Fuel Supply	Fuel Inventory	Weekly
9.	Turbine Steam Stop Control Valves	Closure	Yearly
10.	L.P. Steam Dump System (6 lines)	Closure	Monthly
11.	Service Water System	Each pump starts and operates for 15 minutes (unless already operating)	Monthly
12.	City Water Connections to Charging Pumps and Boric Acid Piping	Temporary connections available and valves operable	R

Table 4.1-3 (Sheet 2 of 2)

13.	RHR Valves 730 and 731	Automatic isolation and interlock action	R*
14.	PORV Block Valves	Operability through 1 complete cycle of full travel	R
15.	PORV Valves	Operability	R
16.	Reactor Vessel Head Vents	Operability	R

R Each Refueling Outage

* If not done during the previous 18 months, the check will be performed next time the plant is cooled down.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 93 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

INTRODUCTION

By letter dated March 10, 1989, as supplemented by letter dated February 9, 1990, the Power Authority of the State of New York (the licensee) proposed to change the Technical Specifications (TS) of Indian Point Unit 3 (IP-3) to (1) reduce the frequency of required testing of the turbine steam stop and control valves, and (2) eliminate the Limiting Conditions for Operation (LCOs) and Surveillance Requirements for the turbine independent electrical overspeed protection system (IEOPS).

It should be noted that the licensee's justification of both proposed TS changes relies on the results in Westinghouse Topical Report WCAP-11525, "Probabilistic Evaluation of Reduction in Turbine Valve Test Frequency," June 1987. That report assumed a maximum turbine overspeed condition of 126%. The licensee stated that the IP-3 licensing basis requires consideration of a maximum overspeed of 132%. Accordingly, the licensee submitted changes to WCAP-11525 which showed a new conditional probability for missile ejection given the occurrence of design overspeed and a new total annual probability of turbine missile ejection due to overspeed. These changes to WCAP-11525 were used by the licensee in providing the data it submitted in support of the proposed TS changes.

The licensee's letter dated February 9, 1990 provided clarifying information and corrected TS pages. None of these changes significantly changed the licensee's application.

EVALUATION OF PROPOSED REDUCTION IN TURBINE VALVE TEST FREQUENCY

On February 7, 1989, the NRC staff issued a generic safety evaluation (SE) which accepts the methodology described in WCAP-11525 for use in TS change proposals generated by those members of the Westinghouse Owners Group (WOG) who participated in the Turbine Valve Test Frequency Evaluation Subgroup (Subgroup). In a subsequent letter dated November 2, 1989, the staff encouraged members of the WOG Subgroup to propose TS changes that:

(1) are consistent with the model TS

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(2) are responsive to the requirements for plant-specific information listed below:

a. if applicable, provide a justification for use of a total turbine missile generation probability other than less than 0.00001 per year.

b. Provide a commitment to work with the turbine vendor to maintain a turbine valve data for the purpose of tracking changes in valve component failure rates.

c. Provide a commitment to accumulate valve failure rate information in a manner accessible for staff audit, and a commitment that this information will be reviewed by the licensee at least once every three years and updated when more than minor changes occur in the data.

d. Provide a commitment to review and reevaluate the turbine valve testing frequency probabilistic analysis (by the methodology of WCAP-11525) when any major changes to the turbine system are made in accordance with 10 CFR 50.59 or when a significant upward trend in the valve failure rate data is identified.

Overall, the effect of allowing the IP-3 licensee to implement the proposed TS change will be to decrease testing of the IP-3 turbine steam stop and control valves from monthly to (a) a frequency which is determined by the methodology shown in WCAP-11525, (b) a frequency which is in accordance with the criteria previously accepted by NRC for the probability for a missile ejection incident at IP-3, and (c) at least annually. The justification for the TS change is that the evaluation presented in WCAP-11525 demonstrates that a significant increase in the interval between turbine valve function tests can be achieved without exceeding the NRC plant-specific acceptance criteria for the probability of a turbine missile accident. The safety benefit for the TS change is that a decreased turbine value test frequency reduces the likelihood of inadvertent reactor trips which can occur during the power change required to perform the valve tests.

The NRC Safety Evaluation dated February 26, 1987, approved an initial value of turbine failure and missile generation probability of 10^{-5}/year for Indian Point 3. Table 1 of the attachment to the Safety Evaluation states that if the probability is between 10^{-5}/year and 10^{-4}/year, the turbine may be kept in service until the next scheduled outage. Indian Point 3 met the 10^{-5}/year criterion based on a three year inspection cycle for LP-1 and LP-3 and a five year inspection cycle for LP-2. The probabilities of an IP-3 turbine missile ejection incident range from 2.54×10^{-5} with a turbine valve inspection interval of one month to 2.62×10^{-5} with a turbine valve inspection interval of one year. These numbers demonstrate that reducing the frequency of turbine valve testing does not result in a significant change in the failure rate and the probabilities are within the NRC acceptance criteria.

The licensee stated upon implementation of these proposed Technical Specification changes, the methodology contained in WCAP-11525 will establish the licensing basis for limiting the total annual probability of turbine missile ejection. Information on valve failure rate will be included in the plant annual final safety analysis report (FSAR) update. The failure rate information included in the FSAR will be updated at least once every three years. The licensee has been and will continue working with the turbine/turbine valve vendor to maintain a turbine valve failure database for the purpose of tracking changes in valve failure rate. The turbine valve testing frequency probabilistic analysis (by the WCAP-11525 methodology) will be reviewed and reevaluated any time major changes in the turbine system have been made or a significant upward trend in the valve failure rate is identified.

The licensee expects to replace the turbine rotors with a superior design during the cycle 7/8 refueling in late 1990. It is anticipated that the new rotors will provide a reduction in missile generation probability of at least an order of magnitude.

Based on our review, we conclude that the proposed changes are acceptable.

EVALUATION OF PROPOSED ELIMINATION OF TS REQUIREMENTS FOR IEOPS

In a letter dated February 2, 1987, to the Westinghouse Electric Corporation, Generation Technology Systems Division (Mr. James A. Martin), the NRC staff stated its belief that maintaining, through testing and inspection, an initial small value of the probability of turbine failure resulting in the ejection of fragments through the turbine casing is a reliable means of ensuring that the objectives precluding turbine missiles and unacceptable damage to safety-related structures, systems, and components can be met. Maintaining an initial small value of the probability of a turbine failure as discussed above simplifies and improves procedures for evaluation of turbine missile risks and ensures that the public health and safety is maintained. To implement this emphasis, the staff proposed, in the letter dated February 2, 1987, turbine failure guidelines for total turbine missile generation probabilities to be used for determining (1) frequencies of turbine disc ultrasonic inspections and (2) maintenance and testing schedules for turbine control and overspeed protection systems. In the letter dated February 2, 1987 to Westinghouse, the NRC issued reliability criteria for maintaining the turbine in service dealing with the turbine missile generation probability less than $1x10^{-4}$ per year for a favorably-oriented turbine and $1x10^{-5}$ per year for an unfavorably-oriented turbine. This provides adequate assurance that the guidelines values of Section 2.2.3 of the Standard Review Plan are satisfied.

The proposed amendment is based in part on the methodology described in WCAP-11525. That methodology was constructed assuming that the IEOPS does not exist. WCAP-11525 concludes, even without taking credit for the IEOPS, that the acceptance criterion for the frequency of turbine missile generation described in the preceding paragraph is satisfied based on the current low pressure turbine rotor inspection intervals and turbine valve test frequency, as well as turbine valve test frequencies up to 12 months.

The dominant faults associated with turbine missile ejection are associated with valve related failures (e.g., steam dump valves fail to open, stop/control valves fail to close). Multiple turbine actuation trains, such as the case with the IEOPS, provide only minimal additional protection since they ultimately actuate the same valves to preclude an overspeed condition.

For the reasons discussed above, the staff has determined that even without relying on the IEOPS the probability of missile generation as a result of overspeed events has been demonstrated to satisfy the applicable turbine missile acceptance criterion. The staff concludes that the proposed amendment to the Indian Point Unit 3 TS to eliminate the limiting conditions for operation and surveillance requirements for the IEOPS is acceptable.

ENVIRONMENTAL CONSIDERATION

This amendment involves a change in a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Sec 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 this amendment.

CONCLUSION

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: March 19, 1990

PRINCIPAL CONTRIBUTORS:

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C. Nichols