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

October 7, 1994

Mr. William J. Cahill, Jr.
Executive Vice President - Nuclear Generation
Power Authority of the State of New York
123 Main Street
White Plains, NY 10601

SUBJECT: ISSUANCE OF AMENDMENT FOR INDIAN POINT NUCLEAR GENERATING UNIT NO. 3 (TAC NO. M88261)

Dear Mr. Cahill:

The Commission has issued the enclosed Amendment No. 154 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 (TSs) in response to your application transmitted by letter dated November 17, 1993, as supplemented August 9, 1994.

The amendment revises the Technical Specifications to incorporate an instrument calibration "allowable value" format instead of the previous "setting limit" format. Instrument settings changed by this TS amendment include:

(1) The overpressure protection system (OPS) actuation curve (TS Figure 3.1.A-3).

(2) The minimum refueling water storage tank (RWST) water volumes and low level alarm settings (specified in TS Section 3.3.A). In addition the RWST level indicating switch calibration frequency (specified in TS Table 4.1-1) has been changed from once every 18 months to once every 6 months.

(3) The control room ammonia and chlorine toxic gas instrument settings (specified in TS Section 3.3.H).

(4) The containment pressure high and high-high engineered safety features instrument settings (specified in TS Table 3.5.1).

(5) The main steam flow engineered safety features instrument settings (specified in TS Table 3.5.1).

In addition, the TS Bases for protective instrumentation limiting safety system settings (specified in TS Section 2.3) have been revised to clarify the description on constants K_1 through K_6 which are used in the overtemperature delta-temperature and overpower delta-temperature settings.

CP1

DFO1

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,

N. F. Conicella

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

Docket No. 50-286

Enclosures: 1. Amendment No. 154 to DPR-64
2. Safety Evaluation

cc w/encls: See next page

cc:

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

Mr. Gerald C. Goldstein
Assistant General Counsel
Power Authority of the State
of New York
1633 Broadway
New York, NY 10019

Mr. Robert G. Schoenberger
First Executive Vice President
and Chief Operating Officer
Power Authority of the State
of New York
123 Main Street
White Plains, NY 10601

Mr. Leslie M. Hill
Resident Manager
Indian Point 3 Nuclear Power Plant
P.O. Box 215
Buchanan, NY 10511

Mr. Peter Kokolakis
Director Nuclear Licensing - PWR
Power Authority of the State
of New York
123 Main Street
White Plains, NY 10601

Ms. Donna Ross
New York State Energy Office
2 Empire State Plaza
16th Floor
Albany, NY 12223

Charles Donaldson, Esquire
Assistant Attorney General
New York Department of Law
120 Broadway
New York, NY 10271

Resident Inspector
Indian Point 3 Nuclear Power Plant
U.S. Nuclear Regulatory Commission
P.O. Box 337
Buchanan, NY 10511

Mr. Charles W. Jackson
Manager, Nuclear Safety and
Licensing
Consolidated Edison Company
of New York, Inc.
Broadway and Bleakley Avenues
Buchanan, NY 10511

Mayor, Village of Buchanan
236 Tate Avenue
Buchanan, NY 10511

Mr. Richard L. Patch, Director
Quality Assurance
Power Authority of the State
of New York
123 Main Street
White Plains, NY 10601

Union of Concerned Scientists
Attn: Mr. Robert D. Pollard
1616 P Street, NW, Suite 310
Washington, DC 20036

DATED: October 7, 1994

AMENDMENT NO. 154 TO FACILITY OPERATING LICENSE NO. DPR-64-INDIAN POINT UNIT 3

Docket File
PUBLIC
PDI-1 Reading
S. Varga, 14/E/4
J. Zwolinski, 013/H/22
L. Marsh
C. Vogan
N. Conicella
OGC
D. Hagan, 3302 MNBB
G. Hill (2), P1-22
C. Grimes, 11/F/23
S. V. Athavale 8/H/3
S. Brewer, 08/E/23
ACRS (10)
OPA
OC/LFDCB
PD plant-specific file
C. Cowgill, Region I

cc: Plant Service list

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. 154
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 November 17, 1993, as supplemented August 9, 1994, 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:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 154, 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 within prior to restart from the current outage.

FOR THE NUCLEAR REGULATORY COMMISSION

LB Marsh

Ledyard B. Marsh, 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 7, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 154

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

Revise Appendix A as follows:

Remove Pages	Insert Pages
2.3-5	2.3-5
3.1-13	3.1-13
3.3-1	3.3-1
3.3-2	3.3-2
3.3-3	3.3-3
3.3-11	3.3-11
3.3-16	3.3-16
3.3-21	3.3-21
3.5-6	3.5-6
3.5-7	3.5-7
3.5-8	3.5-8

Remove Tables	Insert Tables
3.5-1 (sheet 1 of 2)	3.5-1 (sheet 1 of 2)
4.1-1 (sheet 2 of 6)	4.1-1 (sheet 2 of 6)
4.1-1 (sheet 6 of 6)	4.1-1 (sheet 6 of 6)

The source and intermediate range reactor trips do not appear in the specification as these settings are not used in the transient and accident analysis (FSAR Section 14). Both trips provide protection during reactor startup. The former is set at about 10^{+5} counts/sec and the latter at a current proportional to approximately 25% of rated full power.

The high and low pressure reactor trips limit the pressure range in which reactor operation is permitted. The high pressurizer pressure reactor trip is backed up by the pressurizer code safety valves for overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The low pressurizer pressure reactor trip also trips the reactor in the unlikely event of a loss of coolant accident. Its setting limit is consistent with the value assumed in the loss of coolant analysis. [4]

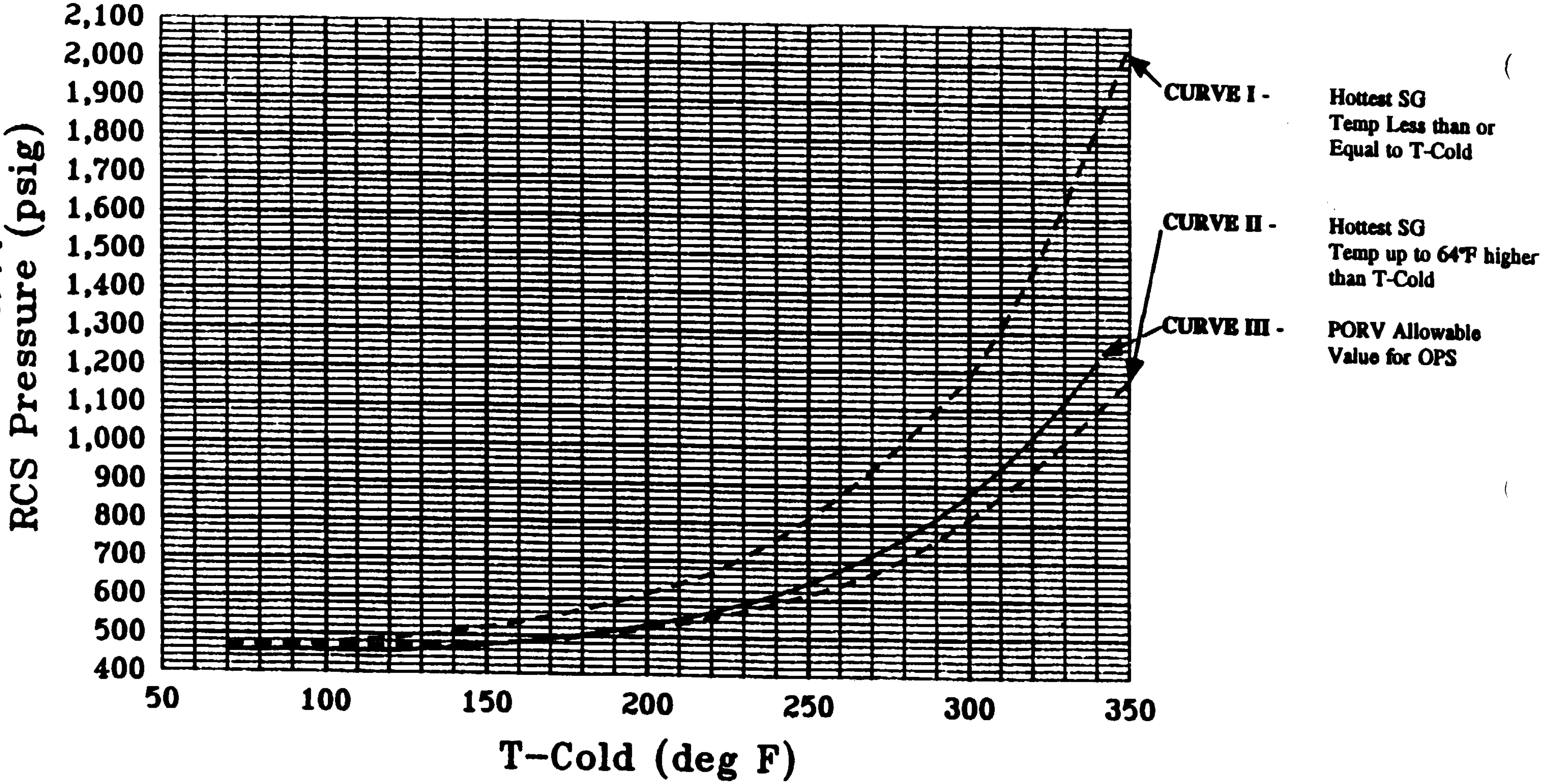
The overtemperature Delta-T reactor trip provides core protection against DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided only that (1) the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 3.5 seconds) [5], and (2) pressure is within the range between the high and low pressure reactor trips. With normal axial power distribution, the reactor trip limit, with allowance for errors [2], is always below the core safety limit as shown on Figure 2.1-1. If axial peaks are greater than design, as indicated by difference between top and bottom power range nuclear detectors, the reactor trip limit is automatically reduced. [6] [7] The values of the constants K_1, K_2, and K_3 are determined during the design of the core for operation with all reactor loops in service. The value for K_1 includes an allowance for instrument channel uncertainty, and therefore is a nominal trip setpoint. K_2 and K_3 are analytical limits, and do not require an allowance for instrument channel uncertainty. The setpoints will ensure that the safety limit of centerline fuel melt will not be reached and the applicable safety limit DNBR will not be violated.

The overpower Delta-T reactor trip prevents power density anywhere in the core from exceeding 118% of design power density, and includes corrections for change in density and heat capacity of water with temperature, and dynamic compensation (via the overall gain in the rate controller) for piping delays from the core to the loop temperature detectors. The specified setpoints meet this requirement. [2] The values of the constants K_4, K_5, and K_6 are determined during the design of the core and the reactor protection system. The value for K_4 includes an allowance for instrument channel uncertainty, and therefore is a nominal trip setpoint. K_5 and K_6 are analytical limits, and do not require an allowance for instrument channel uncertainty.

The overpower limit criteria is that core power be prevented from reaching a value at which fuel pellet centerline melting would occur. Fuel temperature decreases due to cladding creepdown with burnup and consequential reduction of pellet-cladding gap. Thus overpower limits become less restrictive as fuel burnup proceeds.

The T′ values represent the measured full power T_{avg} for the overtemperature and overpower Delta-T equations. T′ must correspond to the indicated full power T_{avg}, and may only be set as high as 573.3°F if the plant operates at the design full power T_{avg}. Reducing T′ for a lower (than design) full power T_{avg} assures that the overtemperature and overpower delta-T setpoint are decreased for any increase in T_{avg} above the indicated loop full power T_{avg}.

Figure 3.1.A-3
RCS Pressure Limits for Low Temperature Operation
Figure applicable to 11 EFPY



Note: Curves I and II include no allowance for instrument error beyond 10 deg F and 30 psi in Appendix–G curve.

Curve III is the OPS PORV allowable value curve.

3.3 ENGINEERED SAFETY FEATURES

Applicability

Applies to the operating status of the Engineered Safety Features.

Objective

To define those limiting conditions for operating that are necessary: 1) to remove decay heat from the core in emergency or normal shutdown situations; 2) to remove heat from containment in normal operating and emergency situations; 3) to remove airborne iodine from the containment atmosphere following a Design Basis Accident; 4) to minimize containment leakage to the environment subsequent to a Design Basis Accident; 5) to minimize the potential for and consequences of Reactor Coolant System pressure transients.

Specification

The following specifications apply except during low temperature physics tests.

A. Safety Injection and Residual Heat Removal Systems

1. The reactor coolant system T_{avg} shall not exceed 200°F unless the following requirements are met:

   a. The refueling water storage tank water level shall be a minimum of 35.4 feet, with the water at a boron concentration $\geq$2400 ppm and $\leq$2600 ppm.

   b. One refueling water storage tank low level alarm operable and set to alarm between 10.5 feet and 12.5 feet of water in the tank.

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 water level shall be a minimum of 35.4 feet, with the 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.

d. One pressure and one level transmitter shall be operating per accumulator.

e. Three safety injection pumps together with their associated piping and valves are operable.

f. Two residual heat removal pumps and heat exchangers together with their associated piping and valves are operable.

g. Two recirculation pumps together with the associated piping and valves are operable.

h. Valves 856B and 856G in the Safety Injection discharge headers shall be closed and their power supplies de-energized.

i. Valve 1810 in the suction line of the high-level SI pumps and valves 882 and 744 in the suction and discharge lines, respectively, of the residual heat removal pumps shall be open and their power supplies de-energized.

j. Valves 842 and 843 in the mini-flow return line from the discharge of the safety injection pumps to the RWST are de-energized in the open position.

k. The refueling water storage tank low level alarms are operable and set to alarm between 10.5 feet and 12.5 feet of water in the tank.

l. Valve 883 in the RHR return line to the RWST is de-energized in the closed position.

m. Valves 1870 and 743 in the miniflow line for the Residual Heat Removal Pumps shall be open and their power supplies de-energized.

4. The requirements of 3.3.A.3 may be modified to allow any one of the following components to be inoperable at any one time:

G. Containment Hydrogen Monitoring Systems

1. One hydrogen monitor including a flow path and associated containment fan cocler unit shall be OPERABLE whenever the reactor T_{avg} exceeds 350°F.

   a. The requirements of 3.3.G.1 can be modified to allow both containment hydrogen monitoring systems to be inoperable for a period not to exceed 7 days.

H. Control Room Ventilation System

1. The control room ventilation system shall be operable at all times when containment integrity is required as per specification 3.6.

2. The requirements of 3.3.H.1 may be modified as follows:

   a. The control room ventilation system may be inoperable for a period not to exceed seventy-two hours. At the end of this period, if the mal-condition in the control room ventilation system has not been corrected, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If after an additional 48 hours the mal-condition still exists, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

3. Two independent toxic gas monitoring systems, with separate channels for detecting chlorine, ammonia, and oxygen shall be operable in accordance with 3.3.H.1 except as specified below. The alarms for ammonia and chlorine shall be adjusted to actuate at ≤35 ppm and ≤3 ppm, respectively.

   a. With any channel for a monitored toxic gas inoperable, restore the inoperable channel to operable status within 7 days.

   b. If 3.a above cannot be satisfied within the specified time, then within the next 8 hours initiate and maintain operation in the control room of alternate monitoring capability for the inoperable channel.

   c. With both channels for a monitored gas inoperable, within 8 hours initiate and maintain operation in the control room of an alternate monitoring system capable of detecting the gas monitored by the inoperable channel.

The minimum indicated RWST level of 35.4 feet (approximately 342,200 gals.), and the low level alarms ("allowable values") of 10.5 feet (approx. 111,100 gals.) and 12.5 feet (approx. 129,700 gals.), include consideration for instrumentation uncertainties, margin, and the unusable volume at the bottom of the tank.[17][18] These water levels ensure a minimum of approx. 195,800 gals. available for injection, and approx. 66,700 gals. for use during and following the transition from injection to recirculation (to allow continued CS pump operation for sump pH control).[18] The minimum RWST boron concentration ensures that the reactor core will remain subcritical during long term recirculation with all control rods fully withdrawn following a postulated large break LOCA.

The four accumulator isolation valves (894 A,B,C,D) are maintained in the open position when the reactor coolant pressure is above 1000 psig to assure flow passage from the accumulators will be available during the injection phases of a loss-of-coolant accident. Indication is also provided on the monitor light panel, should any of these valves not be in the full open position even with the valve operator deenergized. The 1000 psig limit is derived from the minimum pressure requirements of the accumulators combined with instrument error and an operational band and is based upon avoiding inadvertent injection into the reactor coolant system. The accumulator isolation valve motor operators are de-energized to prevent an extremely unlikely spurious closure of these valves from occurring when accumulator core cooling flow is required. Valves 856 B and G are maintained in the closed position to prevent hot leg injection during the injection phase of a loss-of-coolant accident. As an additional assurance of preventing hot leg injection, these valve motor operators are deenergized to prevent spurious opening of these valves during the injection phase of a loss-of-coolant accident. Power will be restored to these valves at an appropriate time in accordance with plant operating procedures after a loss-of-coolant accident in order to establish hot leg recirculation.

Valves 1810, 882, and 744 are maintained in the open position to assure that flow passage from the refueling water storage tank will be available during the injection phase of a loss-of-coolant accident. As additional assurance of flow passage availability, these valve motor operators are de-energized to prevent an extremely unlikely spurious closure. This additional precaution is acceptable, since failure to manually re-establish power to close these valves following the injection phase is tolerable as a single failure.

Valves 842 and 843 in the mini-flow return line from the discharge of the safety injection pumps to the refueling water storage tank are de-energized in the open position to prevent an extremely unlikely spurious closure which would cause the safety injection pumps to overheat if the reactor coolant system pressure is above the shutoff head of the pumps.

These toxic gas monitoring systems are designed to alarm in the control room upon detection of the short term exposure limit (STEL) value. The operability of the toxic gas monitoring systems provides assurance that the control room operators will have adequate time to take protective action in the event of an accidental toxic gas release. Selection of the gases to be monitored are based on the results described in the Indian Point Unit 3 Habitability Study for the Control Room, dated July, 1981. The alarm setpoints will be in accordance with industrial ventilation standards as defined by the American Conference of Governmental Industrial Hygienists.(16)

The OPS has been designed to withstand the effects of the postulated worse case Mass Input (i.e., single safety injection pump) without exceeding the 10 CFR 50, Appendix G curve. Curve III on Figure 3.1.A-3 provides the setpoint curve of the OPS PORVs which is sufficiently below the Appendix G curve such that PORVs overshoots would not exceed the allowable Appendix G pressures. Therefore, only one safety injection pump can be available to feed water into the RCS when the OPS is operable. The other pumps must be prevented from injecting water into the RCS. This may be accomplished, for example, by placing the SI pump switches in the trip pull-out position, or by closing and locking (if manual) or de-energizing (if motor operated) at least one valve in the flow path from these pumps to the RCS. For conditions when the OPS is inoperable, additional restrictions are imposed on the RCS temperature, and pressurizer pressure and level. See Specification 3.1.A.8.b.(3).

<u>References</u>

1) FSAR Section 9
2) FSAR Section 6.2
3) FSAR Section 6.2
4) FSAR Section 6.3
5) FSAR Section 14.3.5
6) FSAR Section 1.2
7) FSAR Section 8.2
8) FSAR Section 9.6.1
9) FSAR Section 14.3
10) FSAR Section 6.8
11) FSAR Section 6.5
12) Response to Question 14.6, FSAR Volume 7
13) FSAR Appendix 14C
14) Response to Question 9.35, FSAR Volume 7
15) WCAP-12313, "Safety Evaluation for an Ultimate Heat Sink Temperature Increased to 95° at IP-3"
16) American Conference of Governmental Industrial Hygienists 1982 Industrial Ventilation, 19th Edition
17) NYPA calculation IP3-CALC-SI-00725, Rev. 0, "Instrument Loop Accuracy/Setpoint Calc./RWST Level."
18) Nuclear Safety Evaluation 93-3-162-SI, Rev. 0, Adequate Post-LOCA Coolant Inventory.

Feedwater Line Isolation

The feedwater lines are isolated upon actuation of the Safety Injection System in order to prevent excessive cooldown of the reactor coolant system. This mitigates the effect of an accident such as steam break which in itself causes excessive coolant temperature cooldown. Feedwater line isolation also reduces the consequences of a steam line break inside the containment, by stopping the entry of feedwater.

Containment Vent and Purge

The containment vent and purge valves are isolated upon actuation of the Safety Injection System, Containment Spray System, or upon receipt of a high containment radiation signal. In the event of an accident, this action prevents a continuous radioactive release via the Containment Vent and Purge System.

Allowable Values

Table 3.5-1 provides the "allowable values" for Engineered Safety Features instrumentation. The "allowable values" represent the limit placed on the "as-found" condition for an instrument loop. If the "as-found" condition measured during calibration is within the "allowable value," the instrument loop will satisfy the system and safety requirements. [6]

1. The Hi-Level containment pressure value is about 10% of containment design pressure. Initiation of Safety Injection protects against loss of coolant[2] or steam line break[3] accidents as discussed in the safety analysis.

2. The Hi-Hi Level containment pressure value is about 50% of containment design pressure. Initiation of Containment Spray and Steam Line Isolation protects against large loss of coolant[2] or steam line break accidents[3] as discussed in the safety analysis.

3. The pressurizer low pressure value is substantially below system operating pressure limits. However, it is sufficiently high to protect against a loss of coolant accident as shown in the safety analysis[2]. The trip is bypassed below 2000 psig to prevent inadvertent actuation of the Engineered Safeguards when the reactor is shutdown.

4. The steam line high differential pressure value is well below those differential pressures expected in the event of a large steam line break accident as shown in the safety analysis[3].

5. The high steam line flow measurement ΔP value is approximately 49% of the full steam flow from no load to 20% load. Between 20% and 100% (full) load, the value for the flow measurement ΔP is ramped linearly with respect to first stage turbine pressure from 49% of the full steam flow to 110% of the full steam flow. High steam flow, coincident with low T_{avg} or low steam line pressure, will initiate safety injection in the case of a large steam line break accident. The coincident low T_{avg} value for SIS and steam line isolation initiation is below the hot shutdown value. The coincident steam line pressure value is below the full load operating pressure. The safety analysis shows that these values provide protection in the event of a large steam line break.[3]

Instrument Operating Conditions

During plant operations, the complete instrumentation systems will normally be in service. Reactor safety is provided by the Reactor Protection System, which automatically initiates appropriate action to prevent exceeding established limits. Safety is not compromised, however, by continuing operation with certain instrumentation channels out of service since provisions were made for this in the Plant design. This specification outlines limiting conditions for operation necessary to preserve the effectiveness of the Reactor Control and Protection System when any one or more of the channels are out of service.

Almost all reactor protection channels are supplied with sufficient redundancy to provide the capability for channel calibration and test at power. Exceptions are backup channels such as reactor coolant pump breakers. The removal of one trip channel on process control equipment is accomplished by placing that channel bistable in a tripped mode; e.g., a two-out-of-three circuit becomes a one-out-of-two circuit. A channel bistable may also be placed in a bypassed mode; e.g., a two-out-of-three circuit becomes a two-out-of-two circuit. The nuclear instrumentation system channels are not intentionally placed in a tripped mode since the test signal is superimposed on the normal detector signal to test at power. Testing of the NIS power range channel requires: (a) bypassing the Dropped Rod protection from NIS, for the channel being tested; and (b) defeating the ΔT protection CHANNEL SET that is being fed from the NIS channel and (c) defeating the power mismatch section of T_{avg} control channels when the appropriate NIS channel is being tested. However, the Rod Position System and remaining NIS channels still provide the dropped-rod protection. Testing does not trip the system unless a trip condition exists in a concurrent channel.

In the event that either the specified Minimum Number of Operable Channels or the Minimum Degree of Redundancy cannot be met, the reactor and the remainder of the plant is placed, utilizing normal operating procedures, in that condition consistent with the loss of protection.

The source range and the intermediate range nuclear instrumentation and the turbine and steam-feedwater flow mismatch trip functions are not required to be operable since they were not used in the transient and safety analysis (FSAR Section 14).

The shunt trip features of the reactor trip and bypass breakers were modified as a result of the Salem ATWS events[4]. Operability requirements for the reactor trip breakers and the reactor protection logic relays were added to the reactor protection instrument operating conditions as a result of NRC review of shunt trip modifications at Westinghouse plants[5]. Operability is demonstrated when the logic coincidence relays are tested to show they are capable of initiating a reactor trip. Reactor trip breakers are considered operable when tested to show they are capable of being opened: (a) by the undervoltage device and the shunt trip device independent of each other from an automatic trip signal and (b) from the Control Room Flight Panel manual trip during refueling outages. An exception of 72 hours is allowed before a reactor trip breaker is declared inoperable if only one of the diverse trip features (undervoltage or shunt trip) fails to open the breaker when tested.

<u>References:</u>

1) FSAR - Section 7.5
2) FSAR - Section 14.3
3) FSAR - Section 14.2.5
4) GL 83-28 - Item 4.3
5) GL 85-09
6) NYPA Report IP3-RPT-MULT-00763, Revision 1, "24 Month Operating Cycle Technical Specification Operability and Acceptance Criteria."

TABLE 3.5-1 (Sheet 1 of 2)

ENGINEERED SAFETY FEATURES INITIATION INSTRUMENT ALLOWABLE VALUES		
No. FUNCTIONAL UNIT	CHANNEL	ALLOWABLE VALUE
1. High Containment Pressure (Hi Level)	Safety Injection	$\leq$ 4.5 psig
2. High Containment Pressure (Hi-Hi Level)	a. Containment Spray b. Steam Line Isolation	$\leq$ 24 psig
3. Pressurizer Low Pressure	Safety Injection	$\geq$ 1700 psig
4. High Differential Pressure Between Steam Lines	Safety Injection	$\leq$ 150 psi
5. High Steam Flow in 2/4 Steam Lines Coincident with Low T_{avg} or Low Steam Line Pressure	a. Safety Injection b. Steam Line Isolation	$\leq$ 6 sec. time delay for SI actuation $\leq$ 49% of full steam flow at zero load $\leq$ 49% of full steam flow at 20% load $\leq$ 110% of full steam flow at full load $\geq$ 540°F T_{avg} $\geq$ 600 psig steam line pressure
6. Steam Generator Water Level (low-low)	Auxiliary Feedwater	$\geq$ 5% of narrow range instrument span each steam generator
7.*a. 480v Bus Undervoltage Relay b. 480v Bus Degraded Voltage Relay (Non-SI) c. 480v Bus Degraded Voltage Relay (Coincident SI)		$\geq$ 200v** $\geq$ 414v with a $\leq$45 sec time delay $\geq$ 414v with a $\leq$10 sec time delay

TABLE 4.1-1 (Sheet 2 of 6)

Channel Description	Check	Calibrate	Test	Remarks
8. 6.9 KV Voltage	N.A.	18M	Q	Reactor protection circuits only
6.9 KV Frequency	N.A.	24M	Q	Reactor protection circuits only
9. Analog Rod Position	S	24M	M	
10. Steam Generator Level	S	24M	Q	
11. Residual Heat Removal Pump Flow	N.A.	24M	N.A.	
12. Boric Acid Tank Level	S	24M	N.A.	Bubbler tube rodded during calibration
13. Refueling Water Storage Tank Level				
a. Transmitter	W	18M	N.A.	Low level alarm
b. Indicating Switch	W	6M	N.A.	Low level alarm
14a. Containment Pressure - narrow range	S	24M	Q	High and High-High
14b. Containment Pressure - wide range	M	18M	N.A.	
15. Process and Area Radiation Monitoring:				
a. Fuel Storage Building Area Radiation Monitor (R-5)	D	24M	Q	
b. Vapor Containment Process Radiation Monitors (R-11 and R-12)	D	24M	Q	
c. Vapor Containment High Radiation Monitors (R-25 and R-26)	D	18M	Q	
d. Wide Range Plant Vent Gas Process Radiation Monitor (R-27)	D	24M	Q	

TABLE 4.1-1 (Sheet 6 of 6)

Table Notation

* By means of the movable incore detector system

** Quarterly 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.

These requirements are applicable when specification 3.3.F.5 is in effect only.

S - Each Shift
W - Weekly
P - Prior to each startup if not done previous week
M - Monthly
NA - Not Applicable
Q - Quarterly
D - Daily
18M - At least once per 18 months
TM - At least every two months on a staggered test basis (i.e., one train per month)
24M - At least once per 24 months
6M - At least once per 6 months

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 154 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 November 17, 1993, as supplemented August 9, 1994, 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 (TSs). The requested changes would revise the TSs to incorporate an instrument calibration "allowable value" format (and associated requirements) instead of the previous "setting limit" format for instruments listed in TS Figure 3.1.A-3, Section 3.3, Table 3.5-1 and Table 4.1-1. In addition, the basis of Section 2.3 would be revised to clarify the description of constants (K_1 through K_6) which are used in equations for the overtemperature delta-temperature (OTDT) and overpower delta-temperature (OPDT) reactor trips settings. The August 9, 1994, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

In January 1993, an audit conducted by the licensee identified a non-compliance with the surveillance test operability criteria for the refueling water storage tank (RWST) low level alarms. This finding was reported to the NRC via Licensee Event Report (LER) 93-009-00, dated March 26, 1993. The licensee concluded from a review of the event that the original methodology used for the RWST low level alarm trip setpoint, allowable value, and TSs limits did not account for all the inaccuracies of the instrument loop, and the surveillance test acceptance criteria did not account for the instrument inaccuracies and drift. Subsequently, the licensee evaluated setpoint calculations and acceptance criteria for all surveillance tests that have a specific alarm, interlock or trip setting, specified in the TSs. As a result of this evaluation, the TSs changes discussed in this safety evaluation were proposed.

2.1 <u>TS Figure 3.1.A-3, Curve III, The Overpressure Protection System (OPS) Actuation Curve.</u>

Reactor pressure vessel integrity at low temperature is protected by opening the Power Operated Relief Valves (PORVs) when the Reactor Coolant System (RCS) pressure reaches a maximum allowable pressure.

This maximum allowable pressure is not a static point but is a function of RCS temperature as shown by the OPS actuation curve. The instrument uncertainty relating to this curve is also a variable function.

<u>Proposed change</u>: The existing OPS actuation curve would be replaced with a new curve. In accordance with TS Section 3.1.A.8.a, this new curve would continue to be used to verify that both OPS PORVs have maximum allowed "as found" lift settings not to exceed the proposed "allowable value" which are shown on the curve. The "allowable value" shown on the existing curve does not represent the true allowable value because it does not include any dynamic margin for instrumentation uncertainties, except a fixed value relating to a temperature 10 °F and pressure 30 psi.

<u>Evaluation</u>: The licensee stated that the proposed new TS curve for "maximum OPS trip setpoint" is based on the loop accuracy calculation for automatic circuitry, and ascertains that the setpoint at any given RCS temperature includes adequate margin to incorporate the instrument uncertainty for that particular point. The new curves are based on a Combustion Engineering (CE) document, MISC-MPS-ER-005, "Pressure-Temperature Limits for IP3," and were drawn according to the following steps.

A. The proposed "analytical OPS limit curve" was drawn by raising each temperature value from the CE document by 50 °F, and reducing each associated pressure value by 70 psi. Raising the temperature was done to compensate for the thermal lag and the pressure was lowered to protect against the design basis pressure overshoot, and to compensate for the response time of the instrument loop and PORVs.

B. On the proposed analytical OPS limit curve, temperature points spaced 10 °F apart were chosen and for each point the total loop uncertainty including drift was calculated. The resulting uncertainties were then converted to an equivalent pressure value. The maximum OPS trip setpoint for each selected temperature point was calculated by subtracting the pressure equivalent of the total loop uncertainty from its analytical value at that point, and all these points were joined by a smooth curve which was the new proposed curve for "maximum OPS trip setpoint."

C. For each temperature point on the proposed analytical OPS limit curve, the "maximum allowable OPS pressure" was computed by adding a pressure equivalent of the drift to the pressure equivalent of the setpoint, and all calculated points were joined by a smooth curve.

The new curves are slightly less restrictive as compared to the existing curves, but the licensee's submittal states that the allowance determined by the revised loop accuracy calculation satisfy the assumptions made in the safety analysis. Because the new OPS curve accounts for loop uncertainty and also satisfies all assumptions and bases of the existing safety evaluation, it is acceptable to the NRC staff.

2.2 <u>Section 3.3 ENGINEERED SAFETY FEATURES, subsection A, Safety Injection and Residual Heat Removal Systems, items 1.a, 1.b, 3.a and 3.k.</u>

<u>Proposed changes</u>: Revise existing sections to read as follows.

Section A.1: The reactor coolant system T_{avg} shall not exceed 200 °F unless the following requirements are met:

a. The RWST water level shall be a minimum of 35.4 feet, with the water at a boron concentration $\geq$2400 ppm and $\leq$2600 ppm;

b. One RWST low level alarm operable and set to alarm between 10.5 feet and 12.5 feet of water in the Tank.

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

a. The RWST water level shall be a minimum of 35.4 feet, with the water at a boron concentration $\geq$2400 ppm and $\leq$2600 ppm;

b. The RWST low level alarms are operable and set to alarm between 10.5 feet and 12.5 feet of water in the tank.

<u>Evaluation</u>: The RWST has two redundant low level alarm circuits to signal the operators to initiate transition from the injection phase to the recirculation phase of the safety injection system. The existing TS requires that the RWST low level alarms be operable and set to alarm between 98,000 gallons and 100,850 gallons of water in the tank prior to the reactor coolant system (RCS) T_{avg} exceeding 350 °F, and one alarm operable prior to the RCS T_{avg} exceeding 200 °F.

The total tank inventory can be divided into three volumes: water needed for recirculation, water in the low level alarm zone, and water needed for injection.

Recirculation Phase: The existing TS requires a minimum of 80,000 gallons for the recirculation phase while the new analysis indicated that 66,700 gallons are required for post LOCA sump PH control, and approximately 40,600 additional gallons are required to account for the minimum water below the instrument tap, unavailable water in the tank bottom, and total of instrument loop uncertainties. Therefore, the total minimum inventory required is equal to 66,700 + 40,600 = 107,300 gallons. For the RWST level monitoring instrumentation loop, the recent setpoint calculations resulted in a total of the loop uncertainty for alarm and indication functions equal to ± 9,300 gallons, and ± 13,000 gallons, respectively. The proposed low level alarm setpoint of 120,400 ± 9,300 gallons (10.5-12.5 feet, proposed change items A.1.b and A.3.k) guarantees minimum of 111,100 gallons (10.5 feet) of water in the RWST which is more than the required inventory of 107,300 gallons. If the accuracy for indication is considered, minimum inventory guaranteed at the alarm setting is 120,400 - 13,000 = 107,400 gallons which is also more than the required minimum of 107,300 gallons. Therefore, the proposed low-level alarm setpoint is acceptable to the NRC staff. With indication accuracy of ± 13,000 gallons at the setpoint of 120,400 gallons, a maximum of 133,400 gallons of water inventory could exist in recirculation volume of the tank.

Injection Phase: The new evaluation determined the total required volume for the injection phase will be 195,800 gallons. This is the total of volumes required to refill the reactor vessel above the nozzles to assure no return to criticality, and the minimum water required in the sump to permit the initiation of recirculation. Adding this volume to the recirculation inventory of 133,400 gallons results in a minimum required RWST volume of 329,200 gallons. Since the indication accuracy was determined to be ± 13,000 gallons, the TS limit for the minimum required inventory equals to 329,200 + 13,000 = 342,200 gallons, which is equal to 35.4 feet of water in the tank. Therefore, the proposed minimum TS level of 35.4 feet (proposed change items A.1.a and A.3.k) will guarantee a minimum of 329,200 gallons for the injection phase and is acceptable to the NRC staff.

2.3 TS Table 4.1-1, MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS, Item 13, Refueling Water Storage Tank Level

Proposed change: Revise Channel Description to include; (a) Transmitter, and (b) Indicating Switch. For Transmitter and Indicating Switch, add "W" in the CHECK column, indicating a requirement to perform a channel check once a week. For Transmitter and Indicating Switch, add "N.A." in the TEST column, indicating no test requirement is required. For Transmitter add "18M" in the CALIBRATE column, indicating a requirement to perform transmitter calibration once every 18 months. For Indicating Switch, add "6M" in the CALIBRATE column, indicating a requirement to perform switch calibration once every 6 months.

Evaluation: The existing Item 13 of TS Table 4.1-1 does not describe the Transmitter and Indicating Switch separately, requires the channel check to be done weekly (W) and a channel calibration to be done once in 18 months, and does not require functional testing. The proposed revision to the TS requires a channel check weekly, transmitter calibration once every 18 months, indicating switch calibration once every 6 months, and does not require any functional testing for either instrument. Therefore, except for changing the calibration frequency of the indicating switch from once every 18 months to once every 6 months, the revised TS does not change any other requirement. The decrease in the alarm calibration frequency for the indicating switch is intended to ensure instrument drift remains within the credit taken in the loop uncertainty calculations. Since the indicating switch calibration frequency is conservative compared to the existing frequency, the proposed TS revision is acceptable to the NRC staff.

2.4 Page 3.3-11, Section H.3, "Control Room Ventilation System"

Proposed change: Revise alarm setpoints for detecting ammonia from "= 35 ppm" to "≤ 35 ppm," and for detecting chlorine from "= 3 ppm" to "≤ 3 ppm."

Evaluation: The proposed change will allow the actual settings in the field to be lower than the TS values. This is conservative and therefore acceptable to the NRC staff.

2.5 Table 3.5-1, ENGINEERED SAFETY FEATURES INITIATION INSTRUMENT ALLOWABLE VALUES, Functional Units 1, 2 and 5

Proposed change: Table 3.5-1, change Title of the third column from "SETTING LIMIT" to "ALLOWABLE VALUE."

For Functional Unit 1, High Containment Pressure (Hi Level), replace the Setting Limit of ≤ 3.5 psig with an allowable value of ≤ 4.5 psig.

For Functional Unit 2, High Containment Pressure (Hi-Hi Level), replace the Setting Limit of ≤ 23 psig with an allowable value of ≤ 24 psig.

For Functional Unit 5, High Steam Flow in 2/4 Steam Lines Coincident with Low T_{avg} or Low Steam Line Pressure,

a. Safety Injection: Change the Setting Limit of ≤40% of full steam flow at zero load to an allowable value of ≤49% of full steam flow at zero load.

b. Steam Line Isolation: Change the Setting Limit of ≤40% of full steam flow at 20% load to an allowable value of ≤49% of full steam flow at 20% load.

<u>Evaluation</u>: The proposed allowable values are the result of the latest setpoint calculations performed by the licensee. The staff reviewed calculations performed by the licensee to evaluate the licensee's calculation methodology. The methodology used by the licensee was based on general guidelines provided by various industry standards, including ANSI/ISA-S67.04-1988, "Setpoints for Nuclear Safety-Related Instrumentation," and USNRC Regulatory Guide 1.105, Revision 2, 1986, "Instrument Setpoints for Safety-Related Systems."

During the initial review of the proposed TS changes, the NRC staff noted that the licensee considered measurement and test equipment (MTE) error, drift, and calibration tolerance as independent terms for both the bistable and sensor, did not provide a specific calibration tolerance (setting tolerance) in the setpoint calculations and elected not to incorporate additional margin in the referenced setpoint methodology or calculation results.

To resolve the NRC staff's initial concerns and provide additional information on the IP3 methodology, the licensee met with the NRC staff on April 14, 1994. The licensee indicated that acceptable limits were being maintained since the allowable values/setpoint methodology used followed the methodology as delineated in ISA standards, Regulatory Guide 1.105, Rev. 2 and past industry practices. Additionally, the licensee stated that based on the use of as-left and as-found data the drift term represents the combined uncertainty due to MTE, drift, and calibration tolerance and reflect a 95% tolerance interval at a 95% confidence level. The licensee also stated that with a reference accuracy greater than or equal to the calibration tolerance and the calibration methodology verifying the channel/instrument performance (to be within the instrument/channel reference accuracy) the calibration tolerance was not included as a setpoint calculation uncertainty term. In addition the licensee stated that for Functional Unit 2 of Table 3.5-1, an even more conservative value will be used as referenced in the licensee's submittal. The existing setting limit values for the other functional units of Table 3.5-1 are conservative with respect to the calculated allowable values and are acceptable to the NRC staff.

The NRC staff indicated that the licensee's methodology may not (in all cases) provide appreciable margin between the analytical limit and the allowable value. As a result, at the end of the calibration interval for both the sensor and transmitter, the allowable value specified by the licensee may not provide adequate margin to the analytical limit such that the inherent accuracy (reference accuracy) of the loop remains bounded. However, based on the above licensee responses and on the licensee's retention of the existing setting limits as allowable values the NRC staff finds the proposed TS changes for Functional Units 1, 2, and 5 acceptable.

2.6 Bases of Section 2.3

Proposed change: Revise to add clarification for description of the constants K_1 through K_6 used in the equations for calculations of OTDT and OPDT setpoints.

Evaluation: The proposed revised text of Bases of Section 2.3 would clarify information relating to constants K_1 through K_6. This is an editorial change and the NRC staff offers no objection to this Bases change.

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 and changes surveillance requirements. 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 67860). 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 Contributors: S. V. Athavale
S. Brewer

Date: October 7, 1994

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,

Original signed by:

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

Docket No. 50-286

Enclosures: 1. Amendment No. 154 to DPR-64
2. Safety Evaluation

cc w/encls: See next page

*See previous concurrence

DOCUMENT NAME: G:\IP3\IP388261.AMD Distribution: See attached sheet

To receive a copy of this document, indicate in the box: "C" = Copy without enclosures "E" = Copy with enclosures "N" = No copy

OFFICE	LA:PDI-1	E	PM:PDI-1	E	OGC *		D:PDI-1	E		
NAME	CVogan		NConicella:avl		RBachmann		LMarsh			
DATE	10/7/94		10/7/94		09/29/94		10/7/94		10/ /94	

OFFICIAL RECORD COPY