BEFORE THE UNITED STATES
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
POWER AUTHORITY OF THE STATE OF NEW YORK	)	Docket No. 50-286
	)	
Indian Point 3 Nuclear Power Plant	)	

APPLICATION FOR AMENDMENT TO OPERATING LICENSE

Pursuant to Section 50.90 of the regulations of the Nuclear Regulatory Commission (NRC), the Power Authority of the State of New York, as holder of Facility Operating License No. DPR-64, hereby applies for an Amendment to the Technical Specifications contained in Appendix A of this license.

The proposed changes to the Indian Point 3 Technical Specifications provide a more suitable alternative to the NRC recommendations in IE Bulletin 79-06A, Item 3, dated April 14, 1979 to the Power Authority.

The proposed changes delete the requirement for placing the low pressurizer level bistable safety injection trip unit of the coincident low pressurizer pressure/level safety injection actuation circuit in the tripped condition which therefore only requires a one of of three low pressurizer pressure signal to generate a safety injection actuation. The proposed changes instead provide for a safety injection actuation signal on a two out of three low pressurizer pressure signal regardless of pressurizer level.

Proposed changes to the Technical Specifications are presented in Attachment I to this Application. The Safety Evaluation is included in Attachment II.

POWER AUTHORITY OF THE
STATE OF NEW YORK

By [signature]
Paul J. Early
Assistant Chief Engineer-
Projects

Subscribed and Sworn to before
me this 26 day of April, 1979.

[signature]
Notary Public

RUTH G. ZAPF
Notary Public, State of New York
No. 30-4663428
Qualified in Nassau County
Commission Expires March 30, 1980

ATTACHMENT I

PROPOSED TECHNICAL SPECIFICATION CHANGES

RELATED TO

LOW PRESSURIZER PRESSURE SAFETY

INJECTION ACTUATION SIGNAL

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
APRIL 18, 1979

3.5 INSTRUMENTATION SYSTEMS

Operational Safety Instrumentation

Applicability

Applies to plant instrumentation systems.

Objectives

To provide for automatic initiation of the Engineered Safety Features in the event that principal process variable limits are exceeded, and to delineate the conditions of the plant instrumentation and safety circuits necessary to ensure reactor safety.

Specification

3.5.1 When the plant is not in the cold shutdown condition, the Engineered Safety Features initiation instrumentation setting limits shall be as stated in Table 3.5-1.

3.5.2 For instrumentation testing or instrumentation channel failure, plant operation shall be permitted to continue in accordance with Tables 3.5-2 through 3.5-4. No more than one channel of a particular protection channel set shall be tested at the same time. By definition, an instrumentation channel failure shall not be regarded as a channel being tested.

3.5.3 In the event the number of in-service channels of a particular function is less than the minimum number of Operable Channels (Col. 3), or the Minimum Degree of Redundancy (Col. 4) cannot be achieved, operation shall be limited according to the requirement shown in Column 5 of Tables 3.5-2 through 3.5-4.

3.5.4 In the event of instrumentation channel failure permitted by specification 3.5.2, the Minimum Degree of Redundancy listed in Tables 3.5-2 through 3.5-4 may be reduced by one, but to not less than zero, and the Minimum Number of Operable Channels listed in these tables may be reduced by one, but not to less than one (except as noted in Table 3.5-3) for a period of 8 hours while instrument channels are tested. The failed channel may be blocked to prevent an unnecessary reactor trip during this time. In the case of three loop operation, the out-of-service channel is permitted to be bypassed during the test period.

3.5.5 The low pressurizer pressure safety injection trip shall be unblocked when the pressurizer pressure is $\geq$ 2000 psig.

3.5.6 At least one source range and one intermediate range nuclear instrument channel shall be operable prior to reactor start-up.

3.5.7 When the reactor is not in the cold shutdown condition, the instrumentation requirements as stated in Table 3.5-5 shall be met.

Basis

Instrumentation has been provided to sense accident conditions and to initiate operation of the Engineered Safety Features[1].

Safety Injection System Actuation

Protection against a Loss of Coolant or Steam Break accident is brought about by automatic actuation of the Safety Injection System which provides emergency cooling and reduction of reactivity.

The Loss of Coolant Accident is characterized by depressurization of the Reactor Coolant System and rapid loss of reactor coolant to the containment. The Engineered Safety Features have been designed to sense the effects of the Loss of Coolant accident by detecting low pressurizer pressure and generate signals actuating the SIS active phase based upon these signals. The SIS active phase is also actuated by a high containment pressure signal (Hi-Level) brought about by loss of high enthalpy coolant to the containment. This actuation signal acts as a backup to the low pressurizer pressure signal actuation of the SIS and also adds diversity to protection against loss of coolant.

Signals are also provided to actuate the SIS upon sensing the effects of a steam line break accident. Therefore, SIS actuation following a steam line break is designed to occur upon sensing high differential steam pressure between any two steam generators or upon sensing high steam line flow in coincidence with low reactor coolant average temperature or low steam line pressure.

The increase in the extraction of RCS heat following a steam line break results in reactor coolant temperature and pressure reduction. For this reason protection against a steam line break accident is also provided by low pressurizer pressure signals actuating safety injection.

Protection is also provided for a steam line break in the containment by actuation of SIS upon sensing high containment pressure.

SIS actuation injects highly borated fluid into the Reactor Coolant System in order to counter the reactivity insertion brought about by cooldown of the reactor coolant which occurs during a steam line break accident.

Containment Spray

The Engineered Safety Features actuation system also initiates containment spray upon sensing a high containment pressure signal (Hi-Hi Level). The containment spray acts to reduce containment pressure in the event of a loss of coolant or steam line break accident inside the containment. The spray cools the containment directly and limits the release of fission products by absorbing iodine should it be released to the containment.

Containment spray is designed to be actuated at a higher containment pressure ($\sim$ 50% of containment design pressure) than the SIS ($\sim$ 10% of containment design pressure). Since spurious actuation of containment spray is to be avoided, it is automatically initiated only on coincidence of Hi-Hi Level containment pressure sensed by both sets of two-out-of-three containment pressure signals and coincidence with the S.I. Signal.

Steam Line Isolation

Steam line isolation signals are initiated by the Engineered Safety Features closing all steam line stop valves. In the event of a steam line break, this action prevents continuous, uncontrolled steam release from more than one steam generator by isolating the steam lines on high containment pressure (Hi-Hi Level) or high steam line flow. Protection is afforded for breaks inside or outside the containment even when it is assumed that there is a single failure in the steam line isolation system.

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.

Setting Limits

1. The Hi-Level containment pressure limit is set at 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 limit is set at 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 limit is set 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 limit is set 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 limit is set at approximately 40% of the full steam flow from no load to 20% load. Between 20% and 100% (full) load, the trip setpoint for the flow measurement ΔP is ramped linearly with respect to first stage turbine pressure from 40% of the full steam flow to 110% of the full steam flow. These setpoints will initiate safety injection in the case of a large steam line break accident. Coincident low Tavg setting limit for SIS and steam line isolation initiation is set below the hot shutdown value. The coincident steam line pressure setting limit is set below the full load operating pressure. The safety analysis shows that these settings 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. 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 Tavg 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).

<u>References</u>:

1) FSAR - Section 7.5
2) FSAR - Section 14.3
3) FSAR - Section 14.2.5

TABLE 3.5-1

ENGINEERED SAFETY FEATURES INITIATION INSTRUMENT SETTING LIMITS

No.	FUNCTIONAL UNIT	CHANNEL	SETTING LIMIT
1.	High Containment Pressure (Hi Level)	Safety Injection	$\leq$ 3.5 psig
2.	High Containment Pressure (Hi-Hi Level)	a. Containment Spray b. Steam Line Isolation	$\leq$ 23 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	$\leq$ 40% of full steam flow at zero load $\leq$ 40% 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

Amendment No.

TABLE 3.5-2 (Sheet 1 of 2)

REACTOR TRIP INSTRUMENTATION LIMITING OPERATING CONDITIONS

		1	2	3	4	5
NO.	FUNCTIONAL UNIT	NO. OF CHANNELS	NO. OF CHANNELS TO TRIP	MIN. NUMBER OF OPERABLE CHANNELS	MIN. DEGREE OF REDUNDNACY	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 $\geq$ 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 $\geq$ 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 2)

		1	2	3	4	5
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 (electrical over-speed protection)	3	2	2	1	Turbine shutdown (turbine stop valves closed)

* 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.

TABLE 3.5-3 (Sheet 1 of 2)

INSTRUMENTATION OPERATING CONDITION FOR ENGINEERED SAFETY FEATURES

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.	SAFETY INJECTION					
	a. Manual	2	1	1	0	Cold Shutdown
	b. High Containment Pressure (Hi Level)	3	2	2	1	Cold Shutdown
	c. High Differential Pressure Between Steam Lines	3/steam line	2/steam line	2/steam line	1/steam line	Cold Shutdown
	d. Pressurizer Low Pressure	3	2	2	1	Cold Shutdown
	e. High Steam Flow in 2/4 Steam Lines Coincident with Low Tavg or Low Steam Line Pressure	2/steam line	1/2 in any 2 steam lines	2 channels in each of 3 steam lines	2	Cold Shutdown or main steam isolation valves closed
		4 Tavg Signals	2	3	2	
		4 Pressure Signals	2	3	2	
	f. Pressurizer Low Pressure and (Automatic Unblock)	3	2	2****	1****	Cold Shutdown

TABLE 3.5-3 (Sheet 2 of 2)

	1	2	3	4	5
2. CONTAINMENT SPRAY					
a. Manual	2	2	2	0***	Cold Shutdown
b. High Containment Pressure (Hi Hi Level)	2 sets of 3	2 of 3 in each set	2 per set	1/set	Cold Shutdown

* Permissible to bypass if reactor coolant pressure less than 2000 psig.

*** Must actuate 2 switches simultaneously.

**** The Minimum Number of Operable Channels and the Minimum Degree of Redundancy may be reduced to zero if the SI bypass is in the unblocked position.

***** If the condition of Column 3 or 4 cannot be met, the reactor shall be placed in the hot shutdown condition, utilizing normal operating procedures, within 4 hours of the occurrence. If the conditions are not met within 24 hours of the occurrence, the reactor shall be placed in the cold shutdown condition, or the alternate condition, if applicable, within an additional 24 hours.

INSTRUMENT OPERATING CONDITIONS FOR ISOLATION FUNCTIONS

NO.	FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MIN. OPERABLE CHANNELS	4 MIN. DEGREE OF REDUN-ANCY	5 OPERATOR ACTION IF CONDITIONS OF COLUMN 3 OR 4 CANNOT BE MET*
1.	CONTAINMENT ISOLATION					
	a. Automatic Safety Injection (Phase A)	See Item No. 1(b) of Table 3.5-3				Cold Shutdown
	b. Containment Pressure (Phase B)	See Item No. 2(b) of Table 3.5-3				Cold Shutdown
	c. Manual					
	Phase A	2	1	1	0	Cold Shutdown
	Phase B	See Item 2(a) of Table 3.5-3				Cold Shutdown
2.	STEAM LINE ISOLATION					
	a. High Steam Flow in 2/4 Steam Lines Coincident with Low T_{avg} or Low Steam Line Pressure	See Item No. 1(e) of Table 3.5-3				Cold Shutdown or Main Steam Isola Valves Closed
	b. High Containment Pressure (Hi Hi Level)	See Item No. 2(b) of Table 3.5-3				Cold Shutdown or Main Steam Isola Valves Closed**
	c. Manual	1/loop	1/loop	1/loop	0	Cold Shutdown or Main Steam Isola Valves Closed

TABLE 3.5-4 (Sheet 2 of 2)

	1	2	3	4	5
3. FEEDWATER LINE ISOLATION					
a. Safety Injection	See Item No. 1 of Table 3.5-3				

* If the conditions of Columns 3 or 4 cannot be met, the reactor shall be placed in the hot shutdown condition, utilizing normal operating procedures, within 4 hours of the occurrence. If the conditions are not met within 24 hours of the occurrence, the reactor shall be placed in the cold shutdown condition, or the alternate condition if applicable, within an additional 24 hours.

** Main steam isolation valves may be closed in lieu of going to cold shutdown if the circuitry associated with closing the valves is the only portion inoperable.

TABLE 3.5-5 (Sheet 1 of 2)

TABLE OF INDICATORS AND/OR RECORDERS AVAILABLE TO THE OPERATOR

		1	2	3
	PARAMETER	NO. OF CHANNELS AVAILABLE	MIN. NO. OF CHANNELS REQUIRED**	INDICATOR/ RECORDER**
1)	Containment Pressure	6	1	Indicator
2)	Refueling Water Storage Tank Level	2	1	Indicator
3)	Steam Generator Water Level (Narrow Range)	3/steam generator	*	Indicator
4)	Steam Generator Water Level (Wide Range)	1/steam generator	*	Recorder
5)	Steam Line Pressure	3/steam line	1/steam line	Indicator
6)	Pressurizer Water Level	3	1	Indicator/One Channel is recorded
7)	RHR Recirculation Flow	4	3	Indicator
8)	Reactor Coolant System Pressure (Wide Range)	1	1	Recorder
9)	Cold Leg Temperature (Tc) (Wide Range)	4	1	Recorder
10)	Hot Leg Temperature (Th) (Wide Range)	4	1	Recorder
11)	Containment Sump Level	2	1	Indicator
12)	Recirculation Sump Level	2	1	Indicator
13)	Temperature Sensors in Penetration Area of Primary Auxiliary Building	3	1	Alarm
14)	Temperature Sensors in Auxiliary Boiler Feedwater Pump Building	2	1	Alarm

	1	2	3
15) Level Sensors in Lower Level of Turbine Building	2	1	Alarm

* One level channel per steam generator (either wide range or narrow range) with at least two wide range channels.

** Columns 2 and 3 may be modified to allow the instrument channels to be inoperable for up to 7 days and/or the recorders to be inoperable for up to 14 days.

If the minimum number of channels required are not restored to meet the above requirements within the time periods specified, 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 Columns 2 and 3 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.

ATTACHMENT II

SAFETY EVALUATION

LOW PRESSURIZER PRESSURE SAFETY

INJECTION ACTUATION SIGNAL

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
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Section I - Description of the Modification

Technical Specification Section 3.5.5, Table 3.5-1 and Table 3.5-3 have been proposed to be modified in order to change the coincident low pressurizer pressure/low pressurizer level safeguard initiation logic to a two out of three initiation logic based on low pressurizer pressure only.

Section II - Purpose of the Modification

These Technical Specification changes are intended to provide an engineered safeguards actuation signal on low pressurizer pressure regardless of pressurizer level.

Section III - Impact of the Change

The existing safeguards logic will initiate a safeguards actuation signal if a low pressurizer pressure condition exists coincident with low pressurizer level. There are three coincident circuits and any one of the three will initiate the actuation. This trip is designed to detect breaks in the reactors coolant system. The coincidence circuits prevents spurious actuation of the Engineered Safeguards System.

An analysis of the events that occurred at Three Mile Island Unit 2 and a review of postulated breaks and valve failures associated with the pressurizer indicate that for certain postulated accidents, the pressurizer level would not decrease coincident with a decrease in pressurizer pressure and thereby not cause an engineered safeguards system actuation.

NRC IE Bulletin 79-06A, dated April 14, 1979 in Item 3 recommended that operating pressurized water reactor power plants which utilized the coincident low pressurizer pressure/level safeguards actuation signal to place the low pressurizer level bistables in the tripped condition. Thus, a one out of three low pressurizer pressure signal would cause an engineered safeguards actuation signal.

The proposed modification would remove the low pressurizer level signal from the safeguards actuation logic and cause an engineered safeguards actuation to occur on low pressurizer pressure, regardless of pressurizer level. This circuit will also prevent spurious actuation of the engineered safeguards system.

The proposed modification to the safeguards actuation logic scheme has been reviewed and found to meet the single-failure criteria of IEEE-279. A change is required, however, to the interlock setpoint for the power operated relief valves. The function of the interlock is to prevent false actuation of the PORV's. PORV actuation occurs on a high pressure signal (2335 psig) to either one of two pre-selected channels whenever a second independent channel satisfies the interlock. The interlock is satisfied whenever the pressure is above the interlock setpoint (presently set at 2185 psig). Adjustment of the interlock setpoint to a value above normal operating pressure of 2235 psig (2285 nominal new setpoint) but below the high pressure PORV reset point will prevent false actuation of the PORV and subsequent blowdown to the pressurizer relief tank due to an assumed failure of the pressure channel (fails high). Thus, a failure in one of the pressure channels will not lead to a condition in which safeguards actuation of the remaining two channels becomes necessary.

A review of the FSAR and SER Accident Analyses indicate that there are no changes to the parameters used, and that the conclusions reached are unchanged, as a result of this modification.

Section IV - Implementation of the Modification

The modifications as proposed will not impact the ALARA or Fire Protection Programs at IP3.

Section V - Conclusions

The incorporation of this modification: a) will not increase the probability nor the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Safety Analysis Report; b) will not increase the possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analyses Report; and c) will not reduce the margin of safety as defined in the basis for any Technical Specification.

Section VI - Reference

(a) IP3 FSAR

(b) IP3 SER