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

May 20, 1994

Docket No. 50-286

Mr. William A. Josiger, Acting
Executive Vice President - Nuclear
Generation
Power Authority of the State of
New York
123 Main Street
White Plains, New York 10601

Dear Mr. Josiger:

SUBJECT: ISSUANCE OF AMENDMENT AND RESCISSION OF CONFIRMATORY ORDER FOR INDIAN POINT NUCLEAR GENERATING UNIT NO. 3 (TAC NO. M85332)

The Commission has issued the enclosed Amendment No. 148 to Facility Operating License No. DPR-64 for the Indian Point Nuclear Generating Unit No. 3. The amendment consists of changes to the Technical Specifications (TS) in response to your application transmitted by letter dated January 6, 1993.

The amendment revises the TS to incorporate the changes listed below:

(1) The residual heat removal (RHR) pump flow calibration frequency (specified in TS Table 4.1-1) has been changed to accommodate operation on a 24-month cycle.

(2) The RHR loop isolation valve automatic isolation and interlock testing frequency (specified in TS Table 4.1-3) has been changed to accommodate operation on a 24-month cycle.

(3) The RHR system leakage testing frequency (specified in TS Section 4.4.I.4) has been changed to accommodate operation on a 24-month cycle.

(4) The recirculation pump testing frequency (specified in TS Section 4.5.B.1.a) has been changed to accommodate operation on a 24-month cycle.

(5) The accumulator check valve operability testing frequency (specified in TS Section 4.5.B.2.b) has been changed to accommodate operation on a 24-month cycle.

(6) The safety injection (SI)/RHR check valve gross leakage testing frequency (specified in TS Section 4.5.B.2.c) has been changed to accommodate operation on a 24-month cycle.

These changes followed the guidance provided in Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," as applicable.

In addition, the gross leakage surveillance requirements for certain SI/RHR system check valves (specified in TS Section 4.5.B.2.d) have been changed to implement recommendations as set forth in NRC Generic Letter, dated February 23, 1980, regarding testing of low pressure injection (LPI)/RHR check valves. Therefore, Item A.5 of the February 11, 1980, Confirmatory Order is considered rescinded.

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

Enclosures:
1. Amendment No. 148 to DPR-64
2. Safety Evaluation

cc w/enclosures:
See next page

RAC 5/4/94

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*Note corrections on page 5.
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These changes followed the guidance provided in Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," as applicable.

In addition, the gross leakage surveillance requirements for certain SI/RHR system check valves (specified in TS Section 4.5.B.2.d) have been changed to implement recommendations as set forth in NRC Generic Letter, dated February 23, 1980, regarding testing of low pressure injection (LPI)/RHR check valves. Therefore, Item A.5 of the February 11, 1980, Confirmatory Order is considered rescinded.

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

Enclosures:
1. Amendment No. 148to DPR-64
2. Safety Evaluation

cc w/enclosures:
See next page

Mr. William A. Josiger
Power Authority of the State
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Indian Point Nuclear Generating
Station Unit No. 3

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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. 148
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 January 6, 1993, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 148, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license is further amended by rescinding Order Item A.5 of Confirmatory Order dated February 1, 1980.

4. This license amendment is effective as of the date of its issuance to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael L. Boyle

Michael L. Boyle, Acting 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: May 20, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 148

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

Revise Appendix A as follows:

Remove Pages	Insert Pages
4.1-3	4.1-3
Table 4.1-1 (Sheet 2 of 6)	Table 4.1-1 (Sheet 2 of 6)
Table 4.1-3 (Sheet 2 of 2)	Table 4.1-3 (Sheet 2 of 2)
4.4-6	4.4-6
4.5-7	4.5-7
4.5-8	4.5-8
4.5-9	4.5-9
4.5-10	4.5-10
4.5-11	4.5-11

It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g. transient conditions or other ongoing surveillance or maintenance activities. It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are performed at each refueling outage and are specified with an 18-month or 24-month surveillance interval. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed on an 18-month or 24-month basis. Likewise, it is not the intent that 24 month surveillances be performed during power operation unless it is consistent with safe plant operation. The limitation of Definition 1.12 is based on engineering judgement and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval. The phrase "at least" associated with a surveillance frequency does not negate the 25% extension allowance of Definition 1.12; instead, it permits the performance of more frequent surveillance activities.

Based on experience in operation of both conventional and nuclear plant systems, when the plant is in operation, the minimum checking frequency of once per shift is deemed adequate for reactor and steam system instrumentation.

Calibration

Calibrations are performed to ensure the presentation and acquisition of accurate information.

The nuclear flux (linear level) channels are calibrated daily against a heat balance standard to account for errors induced by changing rod patterns and core physics parameters.

Other channels are subject only to the "drift" errors induced within the instrumentation itself and, consequently, can tolerate longer intervals between calibration. Process system instrumentation errors induced by drift can be expected to remain within acceptable tolerances if recalibration is performed at intervals of 18 or 24 months.

Substantial calibration shifts within a channel (essentially a channel failure) will be revealed during routine checking and testing procedures.

Thus, minimum calibration frequencies of once-per-day for the nuclear flux (linear level) channels, and 18 or 24 months for the process system channels is considered acceptable.

TABLE 4.1-1 (Sheet 2 of 6)

Channel Description	Check	Calibrate	Test	Remarks
10. Steam Generator Level	S	18M (1) 24M (2)	Q	1) Indicating circuits only 2) Reactor protection circuits only
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	W	18M	N.A.	Low level alarms
14. Containment Pressure	S	18M	Q	High and High-High
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	
e. Main Steam Lines Process Radiation Monitors (R-62A, R-62B, R-62C, and R-62D)	D	24M	Q	
f. Gross Failed Fuel Detectors (R-63A and R-63B)	D	24M	Q	

TABLE 4.1-3 (Sheet 2 of 2)

13. RHR Valves 730 and 731	Automatic isolation and interlock action	24M
14. PORV Block Valves	Operability through 1 complete cycle of full travel	Quarterly (see Note 1)
15. PORV Valves	Operability	24M
16. Reactor Vessel Head Vents	Operability	24M

24M - At least once per 24 months

Note 1. If the block valve is shut due to a leaking or inoperable PORV, Block Valve operability will be checked the next time the plant is in cold shutdown.

I. Residual Heat Removal System

1. Test

a. (1) The portion of the Residual Heat Removal System that is outside the containment shall be tested either by use in normal operation or hydrostatically tested at 350 psig at the interval specified below.

(2) The piping between the residual heat removal pumps suctions and the containment isolation valves in the residual heat removal pump suction line from the containment sump shall be hydrostatically tested at no less than 100 psig at the interval specified below.

b. Visual inspection shall be made for excessive leakage during these tests from components of the system. Any significant leakage shall be measured by collection and weighing or by another equivalent method.

2. Acceptance Criterion

The maximum allowable leakage from the Residual Heat Removal System components located outside of the containment shall not exceed two gallons per hour.

3. Corrective Action

Repairs or isolation shall be made as required to maintain leakage within the acceptance criterion.

4. Test Frequency

Tests of the Residual Heat Removal System shall be conducted at least once per 24 months.

B. Component Tests

1. Pumps

a. The safety injection pumps, residual heat removal pumps, containment spray pumps and the auxiliary component cooling water pumps shall be started at intervals not greater than one month. The recirculation pumps shall be started at least once per 24 months.

b. Acceptable levels of performance shall be that the pumps start, reach their required developed head on recirculation flow, and operate for at least fifteen minutes.

2. Valves

a. Each spray additive valve shall be cycled by operator action with the pumps shut down at least once per 24 months.

b. The accumulator check valves shall be checked for operability at least once per 24 months.

c. The following check valves shall be checked for gross leakage at least once per 24 months:

857A & G	857J	857S & T	897B
857B	857K	857U & W	897C
857C	857L	895A	897D
857D	857M	895B	838A
857E	857N	895C	838B
857F	857P	895D	838C
857H	857Q & R	897A	838D

d. In addition to 4.5.B.2.c,the following check valves shall be checked for gross leakage every time the plant is shut down and the reactor coolant system has been depressurized to 700 psig or less. This gross leakage test shall also be performed following valve maintenance, repair or other work which could unseat these check valves:

838A	895A	897A
838B	895B	897B
838C	895C	897C
838D	895D	897D

Basis

The Safety Injection System and the Containment Spray System are principal plant safeguards that are normally on standby during reactor operation. Complete systems tests cannot be performed when the reactor is operating because a safety injection signal causes reactor trip, main feedwater isolation and containment isolation, and a Containment Spray System test requires the system to be temporarily disabled. The method of assuring operability of these systems is, therefore, to combine systems tests to be performed during plant shutdowns, with more frequent component tests, which can be performed during reactor operation.

The systems tests demonstrate proper automatic operation of the Safety Injection and Containment Spray Systems. With the pumps blocked from starting, a test signal is applied to initiate automatic action and verification made that the components receive the safety injection signal in the proper sequence. The test demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry.[1]

During reactor operation, the instrumentation which is depended on to initiate safety injection and containment spray is generally checked daily and the initiating circuits are tested monthly (in accordance with Specification 4.1). The testing of the analog channel inputs is accomplished in the same manner as for the reactor protection system. The engineered safety features logic system is tested by means of test switches to simulate inputs from the analog channels. The test switches allow actuation of the master relay, while at the same time blocking the slave relays. Verification that the logic is accomplished is indicated by the matrix test light. The slave relay coil circuits are continuously verified by a built-in monitoring circuit. In addition, the active components (pumps and valves) are to be tested monthly to check the operation of the starting circuits and to verify that the pumps are in satisfactory running order. The test interval of one month is based on the judgement that more frequent testing would not significantly increase the reliability (i.e., the probability that the component would operate when required), yet more frequent testing would result in increased wear over a long period of time.

Other systems that are also important to the emergency cooling function are the accumulators, the Component Cooling System, the Service Water System, and the containment fan coolers. The accumulators are a passive safeguard. In accordance with Specification 4.1, the water volume and pressure in the accumulators are checked periodically. The other systems mentioned operate when the reactor is in operation, and by these means are continuously monitored for satisfactory performance.

The charcoal portion of the containment air recirculation system is a passive safeguard which is isolated from the cooling air flow during normal reactor operation. Hence, the charcoal should have a long useful lifetime. The filter frames that house the charcoal are stainless steel and should also last indefinitely. However, the visual inspection specified in Section A.4(a) of this specification will be performed to verify that this is, in fact, the case. The iodine removal efficiency cannot be measured with the filter cells in place. Therefore, at periodic intervals a representative sample of charcoal is to be removed and tested to verify that the efficiencies for removal of methyl iodide are obtained.[(2)] The fuel storage building air treatment system is designed to filter the discharge of the fuel storage building atmosphere to the facility vent during normal conditions. As required by Specifications 3.8.A.12 and 3.8.C.6, the fuel storage building emergency ventilation system must be operable whenever irradiated fuel is being moved. However, if the irradiated fuel has had a continuous 45-day decay period, the fuel storage building emergency ventilation system is not technically necessary, even though the system is required to be operable during all fuel handling operations. The emergency ventilation fan is automatically started upon high radiation signal and since the bypass assembly is sealed by manually operated isolation devices, air flow is directed through the emergency ventilation HEPA filters and charcoal adsorbers.

High efficiency particulate absolute (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of these adsorbers for all emergency air treatment systems. The charcoal adsorbers are installed to reduce the potential release of radio-iodine to the environment. The in-place test results should indicate a system leak tightness of less than or equal to one percent leakage for the charcoal adsorbers and a HEPA efficiency of greater than or equal to 99 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a methyl iodide removal efficiency of greater than or equal to 90 percent on the fuel handling system samples, and greater than or equal to 85 percent on the containment system samples for expected accident conditions. With the efficiencies of the HEPA filters and charcoal adsorbers as specified, further assurance is provided that the resulting doses will be less than the 10 CFR 100 guidelines for the accidents analyzed.

The basis for the toxic gas monitoring system is given in Technical Specification Section 3.3.

The control room air treatment system is designed to filter the control room atmosphere for intake air and/or for recirculation during control room isolation conditions. The control room air treatment system is designed to automatically start upon control room isolation.

High efficiency particulate absolute (HEPA) filters are installed before the charcoal adsorbers to similarly prevent clogging of these adsorbers. The charcoal adsorbers are installed to reduce the potential intake of radio-iodine by control room personnel. The in-place test results should indicate a system leak tightness of less than or equal to one percent leakage for the charcoal adsorbers and a HEPA filter efficiency of greater than or equal to 99 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a methyl iodide removal efficiency of greater than or equal to 90 percent for expected accident conditions.

With the efficiencies of the HEPA filters and charcoal adsorbers as specified, further assurance is provided that the resulting doses will be less than the allowable levels stated in Criterion 19 of the General Design Criteria for Nuclear Power Plants, Appendix A to 10CFR Part 50.

A pressure drop across the combined HEPA filters and charcoal adsorbers of less than or equal to 6.0 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Pressure drop should be determined at least once per operating cycle to show system performance capability. Proper operation of the system fans should also be verified at least every refueling by either direct or indirect measurements.

If results of charcoal tests are unsatisfactory, two additional samples may be tested. If both of these tests are acceptable, the charcoal may be considered satisfactory for use in the plant. Should the charcoal of any of these air filtration systems fail to satisfy the test criteria outlined in this specification, the charcoal beds will be replaced with new charcoal which satisfies the requirements for new charcoal outlined in Regulatory Guide 1.52 (Revision June, 1973).

The hydrogen recombiner system is an engineered safety feature which would be used only following a loss-of-coolant accident to control the hydrogen evolved in the containment. The system is not expected to be needed until approximately 10 days have elapsed following the accident. At this time, the hydrogen concentration in the containment will have reached 3.0% by volume, which is the design concentration for starting the recombiner system.[3] Actual starting of the system will be based upon containment atmosphere sample analysis. The required surveillance testing of each unit will demonstrate the operability of the system. The bi-annual testing of the containment hydrogen monitoring system will demonstrate the availability of this system.

For the eight flow distribution valves (856 A, C, D, E, F, H, J and K), verification of the valve mechanical stop adjustments is performed periodically to provide assurance that the high head safety injection flow distribution is in accordance with flow values assumed in the core cooling analysis.

Gross leakage testing of the reactor coolant system pressure isolation valves and the Low Pressure Injection(LPI)/residual heat removal(RHR)system valves reduces the probability of an inter-system LOCA(4). These tests implement the requirements set forth in NRC generic letter dated February 23, 1980, regarding testing of LPI/RHR system check valves. This amendment provides a basis for the rescission of item A.5. of a Confirmatory Order issued by the Commission to Indian Point 3 in a letter dated, February 11, 1980. To satisfy ALARA requirements, gross leakage (>10 gpm) may be measured indirectly (i.e. using installed pressure and flow indications).

<u>References</u>

(1) FSAR Section 6.2

(2) FSAR Section 6.4

(3) FSAR Section 6.8

(4) WASH 1400

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 148 TO FACILITY OPERATING LICENSE NO. DPR-64

AND RECESSION OF CONFIRMATORY ORDER

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 January 6, 1993, the Power Authority of the State of New York (the licensee) submitted a request for changes to the Indian Point Nuclear Generating Unit No. 3 (IP3), Technical Specifications (TS). The requested changes would revise the TS to incorporate the following residual heat removal (RHR) system and safety injection (SI) system changes:

(1) The RHR pump flow calibration frequency (specified in TS Table 4.1-1) would be changed to accommodate operation on a 24-month cycle.

(2) The RHR loop isolation valve automatic isolation and interlock testing frequency (specified in TS Table 4.1-3) would be changed to accommodate operation on a 24-month cycle.

(3) The RHR system leakage testing frequency (specified in TS Section 4.4.I.4) would be changed to accommodate operation on a 24-month cycle.

(4) The recirculation pump testing frequency (specified in TS Section 4.5.B.1.a) would be changed to accommodate operation on a 24-month cycle.

(5) The accumulator check valve operability testing frequency (specified in TS Section 4.5.B.2.b) would be changed to accommodate operation on a 24-month cycle.

(6) The SI/RHR check valve gross leakage testing frequency (specified in TS Section 4.5.B.2.c) would be changed to accommodate operation on a 24-month cycle.

The requested changes are needed to accommodate operation on a 24-month fuel cycle. The licensee commenced operating on a 24-month fuel cycle, instead of the previous 18-month fuel cycle, with fuel cycle 9. Fuel cycle 9 started in August 1992. The proposed changes follow the guidance provided in Generic Letter (GL) 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," as applicable.

The licensee also requested that the boron injection tank (BIT) return flow calibration frequency (specified in TS Table 4.1-1) be changed to accommodate operation on a 24-month cycle. However, TS Amendment No. 139, issued on October 15, 1993, allowed the BIT and its associated support systems to be removed. Specifically, the BIT return flow indicator instrument was deleted from TS Table 4.1-1. Therefore, the licensee's request to extend the calibration frequency of this instrument is no longer applicable and was not addressed in this safety evaluation.

In addition, the gross leakage surveillance requirements for certain SI/RHR system check valves (specified in TS Section 4.5.B.2.d) would be changed to implement recommendations as set forth in NRC generic letter dated February 23, 1980, regarding testing of low pressure injection (LPI)/RHR check valves. This would form the basis for rescission of Item A.5 of the February 11, 1980, Confirmatory Order.

2.0 EVALUATION

The licensee considered the following factors in evaluating the emergency core cooling system (ECCS) surveillance interval extensions from 18 to 24 months:

- Does on-line testing adequately demonstrate operability or are failures only being detected during these refueling tests?
- Did past equipment performance have an effect on system safety functions?
- Does performing the surveillance test at power present an unacceptable burden?

The ECCS is an integrated set of subsystems that perform emergency cooling injection and recirculation functions following a loss-of-coolant accident (LOCA). The coolant injection function is performed during a relatively short-term period after LOCA initiation, followed by realignment to a recirculation mode of operation to maintain long-term, post-LOCA core cooling. The ECCS consists of the following subsystems: passive accumulators (one for each loop), safety injection (three high-head pumps), RHR (two low-head pumps and two heat exchangers), and recirculation (two low-head pumps).

2.1 RHR Pump Flow Calibration

During the injection phase of a LOCA, the RHR pumps perform the low pressure injection function. The RHR pumps take suction on the refueling water storage tank (RWST) and inject into all four reactor coolant system (RCS) cold legs. After the injection phase, coolant spilled from the RCS break collects in the containment sump. The contents of the containment sump are then cooled and returned to the RCS by the recirculation system. When RWST level drops to a prescribed low level setpoint, the low pressure injection is realigned so that the recirculation pumps take suction on the containment sump and inject to the RCS cold or hot legs (low-head recirculation). For a small break in the RCS, where recirculated water must be injected against higher pressures for long-term cooling, the safety injection pumps may be aligned to augment the recirculation pumps (high-head recirculation).

The purpose of the RHR pump flow instrumentation calibration is to calibrate the RHR system flow transmitters, associated bistables, and flow indicators. Process flow is monitored through each RHR heat exchanger (FT-638 and 640) and through each injection/recirculation path (FT-946A,B,C,D). The indications provided by this instrumentation are used to verify RHR pump flow during the initial injection phase of a LOCA, to verify total recirculation flow and proper recirculation pump operation during the recirculation phase of a LOCA, to indicate a possible line blockage or break, and to determine whether low-head or high-head recirculation is required.

The licensee reviewed data from 1985 to 1992 related to RHR flow instrumentation calibration. The data indicated that, in general, the instrument calibration results were well within the instrument calibration tolerances and/or vendor's drift allowances. The licensee performed an instrument drift analysis for the flow transmitters, bistables, and indicators to evaluate the acceptability of extending the calibration interval to 30 months (24 months with a 25% extension). The analysis indicated that the maximum expected instrument inaccuracy for 30 months would not affect the instrument's ability to perform its safety function. Therefore, the licensee concluded that this surveillance test interval could be extended since the results of the refueling tests were, in general, satisfactory and the instrument drift analysis justified the extension.

The NRC staff has reviewed the information presented by the licensee regarding RHR flow instrumentation calibration and concludes the requested change is acceptable.

2.2 RHR Automatic Isolation and Interlock Testing

In addition to performing the low-head ECCS function during the injection phase of a LOCA, the RHR system provides shutdown cooling of the RCS after the RCS has been depressurized to less than 400 psig and cooled to less than 350 °F. The RHR system consists of one loop which is made up of two 100% capacity pumps and heat exchangers as well as suction and discharge piping with isolation valves. The RHR suction line is connected to RCS loop 2 hot leg and the return lines connect to all four cold legs.

The RHR system suction path is isolated from the RCS by two in-series motor operated valves (MOVs 730 and 731). Since the RHR system components are designed for lower pressures and temperatures than the RCS, an interlock and automatic isolation feature is provided for these valves. Specifically, the suction valves cannot be opened unless RCS pressure is below RHR design pressure and the valves will automatically close if RCS pressure increases above the RHR design pressure. The purpose of the surveillance test is to verify the operability of the interlock and automatic isolation feature of MOVs 730 and 731. This surveillance test can only be performed during cold shutdown conditions.

The licensee reviewed data from 1987 to 1992 related to RHR interlock and automatic isolation valve testing. The data indicated that all the tests were satisfactory and there was no case where the interlock or the valves would have failed to provide isolation. Therefore, the licensee concluded that this surveillance test interval could be safely extended to a 24-month refueling periodicity. In addition, the 24-month periodicity is consistent with the requirements of NUREG-1431, "Standard Technical Specifications for Westinghouse Plants" (W-STS).

The licensee also proposed deletion of the TS Note which required this surveillance test to be performed the next time the plant was cooled down if the test had not been performed during the previous 18 months. The licensee stated that since the MOVs and interlocks have a good past performance record and the W-STS does not require the provision as stated in the Note, the Note is no longer necessary. The W-STS allows this functional test to be performed each refueling outage. The basis for this W-STS interval is that the plant conditions needed to perform this surveillance occur during an outage. This frequency is also acceptable based on the design reliability and confirming operating experience of the MOVs and interlocks.

The NRC staff has reviewed the information presented by the licensee regarding RHR automatic isolation and interlock testing concludes the requested changes are acceptable.

2.3 RHR System Leakage Testing

The external recirculation flow paths (RHR and safety injection) are tested to ensure total system leakage does not exceed 2 gallons per hour. This will limit off-site exposures, due to leakage, to insignificant levels relative to those calculated for leakage directly from the containment in the design basis accident. During the test, all associated system joints, valve packing, pump seals, leakoff connections and other potential points of leakage are visually examined. The leakage testing can only be performed with the plant shutdown and the RHR system in service.

The licensee reviewed data from 1987 to 1992 related to RHR system leakage testing. The data indicated that in all the tests, except for one, system leakage was well below the 2 gallons per hour limit. The one test that failed had leakage slightly greater than the limit. However, this leakage was attributed to a safety injection pump seal and was detected during the monthly pump run. The licensee indicated that during plant operation, the likelihood of developing leaks while the system was in standby was minimal. Extending the surveillance interval would only extend the period when the system is in standby. Therefore, based on the good surveillance history and the low likelihood of increased leakage, the licensee concluded that this surveillance test interval could be safely extended to a 24-month refueling periodicity.

The NRC staff has reviewed the information presented by the licensee regarding RHR system leakage testing and concludes the requested changes are acceptable.

2.4 Recirculation Pump Testing

After the injection phase of a LOCA, coolant spilled from the break collects in the recirculation sump, which is located in the containment floor, is cooled and returned to the RCS by the recirculation system. The recirculation system consists of two 100% capacity pumps and suction piping. The recirculation pumps discharge to the RHR heat exchangers and from this point on, the low-head RHR injection piping is used. The complete recirculation system is located internal to the containment. The RHR pumps can provide backup to the internal recirculation system; however, the RHR recirculation loop would be external to the containment and the RHR recirculation suction would be from the containment sump.

The purpose of the refueling surveillance is to demonstrate the operability of the recirculation pumps and to exercise each pump's discharge check valves. The refueling periodicity is consistent with the inservice testing (IST) requirements of Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. Testing these components with the reactor at power is impractical since the recirculation sump is dry. The refueling outage test involves filling the recirculation sumps with 5000 gallons of water in order to operate the pumps then, disposing of the 5000 gallons of contaminated waste water. Thus, the effort required to prepare for and restore from this test makes a test frequency other than a refueling frequency impractical. The licensee also stated that since these pumps stand idle and dry except for periods of testing, significant inservice degradation is unlikely.

The licensee reviewed data from 1987 to 1992 related to recirculation pump testing. The data indicated that all the tests were satisfactory for the discharge check valves and there was only one recirculation pump failure. However, the pump failure was a retest after a modification had been improper installed. Therefore, based on the good surveillance history and the availability of the redundant RHR external recirculation system, the licensee concluded that this surveillance test interval could be safely extended to a 24-month refueling periodicity.

The NRC staff has reviewed the information presented by the licensee regarding recirculation pump testing and concludes the requested changes are acceptable.

2.5 Accumulator Check Valve Operability Testing

The accumulators are filled with borated water and are pressurized to about 650 psig with nitrogen. Following a LOCA the four accumulators (one for each RCS cold leg) will discharge its contents into the RCS when RCS pressure falls below accumulator pressure, thereby reflooding the core. During normal power operation, the accumulators are isolated from the RCS by check valves 895 A,B,C,D. Therefore, mechanical action of these swing-disc check valves is the only action required to open the injection path.

The licensee's current TS requires that these check valves to be tested for operability during each refueling outage. This testing is also required by the licensee's IST program for pumps and valves. This operability test involves exercising each valve to the open position. This testing cannot be done during power operation since accumulator pressure is much less than normal RCS pressure. Specifically, during each refueling outage, each of these check valves are part-stroke exercised and a leakage test is performed to verify closure. In addition, in accordance with the guidance of NRC GL 89-04, "Guidance on Developing Acceptable Inservice Testing Programs," a sample disassembly and inspection of these valves is performed to verify full-stroke capability.

The licensee reviewed data from 1985 to 1992 related to accumulator check valve testing. The data indicated that all tests were satisfactory. Work requests that were issued were for preventive maintenance and check valve disassembly for inspection. In 1989, the retaining block studs for all four accumulator check valves, which are Anchor Darling check valves, were replaced with type A564 stainless steel material in accordance with NRC Information Notice 88-85, "Broken Retaining Block Studs on Anchor Darling Check Valves." Based on the results of the review, the licensee concluded that this surveillance test interval could be extended since the accumulator check valves have proven to be reliable and there was no evidence that check valve performance was a function of the surveillance interval.

The NRC staff has reviewed the information presented by the licensee regarding accumulator check valve testing and concludes the requested changes are acceptable.

2.6 <u>SI/RHR Check Valve Leakage Testing</u>

Gross leakage testing is performed on safety injection (SI) system check valves 857A-W, 895A-D, 897A-D, and 838A-D to verify proper valve closure and leak tightness. The proposed TS amendment would change the required frequency for gross leakage testing for check valves from 18 months to 24 months to accommodate a 24-month refueling cycle.

In addition to the TS required gross leakage testing, the valves are part-stroke exercised and a sample of valves are disassembled and inspected once per refueling outage to demonstrate full-stroke capability as part of the actions taken in response to Generic Letter 89-04, "Guidance on Developing Acceptable Inservice Testing Programs," dated April 3, 1989. These tests provide additional assurance that the valves will work when called upon.

The licensee reviewed surveillance tests results from 1985 through 1992 (897s, 838s, and 895s) and from 1988 through 1993 (857s). The tests results indicated that all the subject valves passed the surveillance tests satisfactorily. The licensee indicated in their response, and in previously NRC approved American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, relief requests, that these valves are not testable at power. By letter dated November 9, 1993, Revision 4 of the

IP3 Second 10-Year Interval Inservice Testing (IST) Program was submitted to the NRC for review and approval of various ASME Code relief requests. These SI system check valves are all required to be leak tested per the licensee's IST program; however, the 24-month testing frequency is consistent with ASME Code requirements. Therefore, relief from the requirement of the ASME Code is not needed for leak testing these check valves on a 24-month frequency.

The NRC staff has reviewed the information presented by the licensee regarding the testing of the subject check valves and concludes the requested change is acceptable provided the valves remain IST Category A/C valves and they continue to be tested for leakage as currently required by the licensee's IST program.

2.7 Rescission of Confirmatory Order

In late 1979, based on the belief that the Indian Point power plants presented a disproportionately high contribution to the total societal risk from reactor accidents due to the relatively high population density surrounding the Indian Point site as compared to other nuclear power plant sites, a program was undertaken to study, evaluate, and implement potential permanent plant modifications and emergency procedures that would reduce the probability and consequences of such an accident. The program was to use probabilistic risk assessment techniques. By letter dated February 1, 1980, the licensee committed to a number of interim and long-term actions that would be taken to reduce the already low risk. One of these interim actions was to conduct special testing of the LPI/RHR check valves to ensure they functioned properly as pressure isolation devices. These affected valves included SI check valves 838A-D, 895A-D, and 897A-D and the special testing involved assuring that the subject check valves were in fact installed correctly and functioning as pressure isolation barriers when the plant was at power and verifying check valve operability whenever RCS pressure decreased to within 100 psig of RHR system pressure. In addition TS Section 4.5.B.2.d required these valves to be checked for gross leakage midway between refuelings. By letter dated February 11, 1980, the NRC issued a Confirmatory Order which, in part, confirmed by Order the licensee's commitments as stated in the February 1, 1980, letter. The LPI/RHR check valve testing was Order Item A.5.

On February 23, 1980, the NRC issued GL 80-14, "LWR Primary Coolant System Pressure Isolation Valves." The generic letter requested all licensees to identify high pressure/low pressure interfaces which could result in an intersystem LOCA and to assure component integrity. By letter dated March 13, 1980, the IP3 licensee responded to the generic letter and indicated that Confirmatory Order Item A.5 was being conducted and this item addressed the concerns of the generic letter. In 1981, valve operability and leakage testing recommendations, as set forth in the February 23, 1980, generic letter, were imposed on most licensees other than the IP3 licensee by Orders.

By letter dated July 5, 1985, the NRC issued a Rescission of Order which rescinded the February 11, 1980, Confirmatory Order. The rescission was based on Commission Decision CLI-85-06, dated May 7, 1985. However, several items of the Order were generic issues and these items would be rescinded when compliance with the generic issue was confirmed. One such item was Item A.5.

Specifically, the Rescission of Order stated that Item A.5 was to remain in effect until a TS amendment implementing the recommendations of the February 23, 1980, generic letter was approved by the NRC staff. Upon such approval, the Order would be rescinded with respect to Item A.5.

As previously stated, the TS Section 4.5.B.2.d currently requires testing for gross leakage midway between refuelings. The licensee has proposed modifying this TS to add the requirements for the subject check valves to be checked for gross leakage every time the plant is shut down and the RCS has been depressurized to 700 psig (i.e. within 100 psig of RHR system pressure). In addition the TS would require gross leakage test following maintenance, repair or other work which could unseat these valves. The gross leakage testing is performed to verify proper valve closure and leak tightness.

The NRC staff has reviewed the information presented by the licensee regarding leakage testing of the RCS pressure isolation valves, located in the SI system, and concludes the requested change is acceptable provided the valves remain IST Category A/C valves and they continue to be tested for leakage as currently required by the licensee's IST Program. Since this TS change incorporates the intent of GL 80-14, the February 11, 1980, Confirmatory Order Item A.5 is considered rescinded.

2.8 Summary

The licensee has evaluated the effect of the increase in the surveillance interval on safety for each of the proposed changes and has concluded that the effect is small. The licensee has confirmed that historical plant maintenance and surveillance data do not invalidate this conclusion. The increase in each of the surveillance intervals to accommodate a 24-month fuel cycle does not invalidate any assumption in the IP3 licensing basis.

The staff has reviewed the information presented by the licensee and concludes that the proposed changes do not have a significant effect on safety and they are consistent with the W-STS or follow the guidance of GL 91-04, as applicable. Therefore, all the proposed changes are acceptable. In addition, Item A.5 of the February 11, 1980, Confirmatory Order is considered rescinded.

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 8777). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor:
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DATED: May 20, 1994

AMENDMENT NO. 148 TO FACILITY OPERATING LICENSE NO. DPR-64-INDIAN POINT UNIT NO. 3

Docket File
NRC & Local PDRs
PDI-1 Reading
S. Varga, 14/E/4
J. Calvo, 14/A/4
M. Boyle
C. Vogan
N. Conicella
OGC
D. Hagan, 3302 MNBB
G. Hill (2), P1-22
Wanda Jones, P-370
C. Grimes, 11/F/23
ACRS (10)
OPA
OC/LFDCB
PD plant-specific file
C. Cowgill, Region I
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cc: Plant Service list