ATTACHMENT I TO IPN-93-001

PROPOSED TECHNICAL SPECIFICATION CHANGES

RELATED TO

SAFETY INJECTION AND RESIDUAL HEAT REMOVAL SYSTEM TESTING

AND 24 MONTH OPERATING CYCLES

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
DPR-64

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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.

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 5)

Channel Description	Check	Calibrate	Test	Remarks
10. Steam Generator Level	S	18M	Q	
11. Residual Heat Removal Pump Flow	N.A.	24M	N.A.	
12. Boric Acid Tank Level	S	18M	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 Systems	D	18M	Q	
16. Containment Water Level Monitoring System:				
a. Containment Sump	N.A.	18M	N.A.	
b. Recirculation Sump	N.A.	18M	N.A.	Narrow Range, Analog
c. Containment Water Level	N.A.	18M	N.A.	Narrow Range, Analog Wide Range
17. Accumulator Level and Pressure	S***	18M	N.A.	
18. Steam Line Pressure	S	18M	Q	
19. Turbine First Stage Pressure	S	18M	Q	
20. Reactor Protection Relay Logic	N.A.	N.A.	TM	
21. Turbine Trip Low Auto Stop Oil Pressure	N.A.	18M	N.A.	
22. Boron Injection Tank Return Flow	S	24M	N.A.	

Amendment No. 8, 38, 63, 68, 74, 93, 107, 123,

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	18M
15.	PORV Valves	Operability	18M
16.	Reactor Vessel Head Vents	Operability	18M

18M - At least once per 18 months
24M - At least once per 24 months

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 18 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

Amendment No. 123,

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).

References

(1) FSAR Section 6.2

(2) FSAR Section 6.4

(3) FSAR Section 6.8

(4) WASH 1400

ATTACHMENT II TO IPN-93-001

SAFETY EVALUATION

RELATED TO

SAFETY INJECTION AND RESIDUAL HEAT REMOVAL SYSTEM TESTING

AND 24 MONTH OPERATING CYCLES

TECHNICAL SPECIFICATION CHANGE

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
DPR-64

Section I - Description of Changes

This application for amendment to the Indian Point 3 Technical Specifications proposes to change the frequency of safety injection (SI) and residual heat removal (RHR) system testing to accommodate operation with a 24 month operating cycle.

Starting with cycle nine (that began in August, 1992), Indian Point 3 began operating on 24 month cycles, instead of the previous 18 month cycles. The specific Technical Specifications that will be changed by this application are:

- Residual Heat Removal Pump Flow Calibration
- Boron Injection Tank Return Flow Calibration
- Residual Heat Removal Loop Isolation Valve (730, 731) Interlock Test
- Residual Heat Removal System Leakage Test
- Recirculation Pump Test
- Safety Injection/Residual Heat Removal System Check Valve Operability Tests
- Basis page 4.1-3 to include discussion of 24 month frequency

This application also proposes to amend the surveillance requirement of certain SI/ RHR system check valves to implement the requirements set forth in NRC generic letter dated February 23, 1980, regarding testing of LPI/RHR check valves.

Section II - Evaluation of Changes

Starting with cycle nine (that began in August, 1992), Indian Point 3 began operating on 24 month cycles, instead of the previous 18 month cycles. To avoid either a surveillance outage at 18 month intervals or an extended mid-cycle outage, changes are required to system surveillance test intervals. In evaluating the extension of surveillance intervals to be consistent with the length of the operating cycle, the following factors were considered: the importance of the refueling tests (i.e., does on-line testing demonstrate operability, or are failures only being detected during the refueling tests?), past equipment performance (and the effect on system safety functions), and the burden of performing tests during power operation.

The Authority has been doing check valve testing to satisfy the requirements of item A.5. of a Confirmatory Order issued by the Commission to Indian Point 3 in a letter dated, February 11, 1980. The Authority is proposing a Technical Specification Change to implement the requirements set forth in NRC generic letter dated February 23, 1980, regarding testing of LPI/RHR check valves to provide a basis for the rescission of this order.

Below is an evaluation for each technical specification that this application proposes to change.

Residual Heat Removal (RHR) Pump Flow Calibration

The purpose of this test is to calibrate the RHR system flow transmitters, associated bistables and flow indicators (FI-946A, B, C, and D), in accordance with Indian Point 3 Technical Specification Table 4.1-1, item 11. These flow transmitters monitor the process flow through the RHR heat exchangers (FT-638 and 640) and the reactor coolant system (RCS) cold leg

injection/recirculation path (FT-946A, B, C and D) and provide indication of its corresponding system status in the main control room. The operator uses these indications to: 1) verify RHR pump flow during initial phase of safety injection, and 2) verify the total recirculation flow and proper RHR pump operation during the recirculation phase of safety injection. An RHR low flow alarm is provided via bistables FC-946A, B, C, and D to alert the operator of a possible blockage or line break. Also, the operator depends on FI-946A, B, C, and D to determine if low-head or hi-head recirculation is required during the recirculation phase of safety injection.

A review of the surveillance test results and operating occurrence reports from 1985 through mid-1992 revealed that past performance of FT-638, 640, 946B, and 946D, FC-946A, B, C, and D were all satisfactory and the instrument calibration results were well within the instrument calibration tolerances and/or vendor's drift allowances. However, the calibration results for FT-946A and FT-946C were out of specification during the 6/87 and 5/92 tests, respectively, and flow indicators FI-638 and FI-640 were out of specification during the 1990 test.

An instrument drift analysis for the flow transmitters, bistables and indicators was performed to evaluate the acceptability of extending the calibration interval from a maximum of 22.5 months to a maximum of 30 months (24 months plus a 25% tolerance) corresponding to the 24 month operating cycle. This analysis concluded that RHR flow calibrations could be safely extended because the maximum expected instrument inaccuracy for 30 months will not affect the instrument's ability to perform its safety function.

Boron Injection Tank (BIT) Return Flow Calibration

The purpose of this test is to verify the calibration of the boron injection tank recirculation flow indicator, in accordance with IP-3 Technical Specification Table 4.1-1 Item 22. The level in the BIT is maintained at 100% by continuously recirculating the water to the boric acid storage tanks (BAST's). The boric acid transfer pumps are used to circulate the solution between the BIT and both BASTs, and the flow is verified periodically on FI-916. The plant must be in the cold shutdown condition, as specified in IP-3 Technical Specification Section 3.2.B, to perform this test.

A review of the surveillance test results from 1987 to mid-1992 indicated that past test results (average flow indications of the flow indicator) agreed with the calculated flow rate within calibration tolerance of ± 10 gpm, and were considered operable and acceptable.

An instrument drift analysis for the flow indicator was also performed to evaluate the acceptability of extending the calibration interval from a maximum of 22.5 month to a maximum of 30 months. Results of this analysis concluded that the 30 month best estimate drift (BED30) is well within the calibration tolerance. The 30 month best estimate of drift was determined as follows: (1) the field drift (as found value minus the previous interval's as left value) data points are determined; (2) the drift values from step (1) were extrapolated to 30 month values using the square root of the sum of the squares technique; (3) finally the 30 month values were arithmetically averaged, to give a "best estimate of drift" for a 30 month period.

Based on its past good calibration record, and favorable result of the instrument drift analysis, this surveillance test interval can be safely extended to 24 months.

Residual Heat Removal Loop Isolation Valve (730,731) Interlock Test

The purpose of this test is to demonstrate the operability of the interlocks associated with RHRS valves, 730 and 731 and the isolation capability of the valves themselves. The test requirements are in accordance with Technical Specifications Table 4.1-3, item 13, which requires that interlocks associated with RHR valves 730 and 731 to be tested for automatic isolation and interlock action every refueling outage. If the test is not done during the previous 18 months, this test will be performed next time the plant is cooled down.

A review of the past surveillance test results from 7/87 through 6/92 indicated that all test results were satisfactory. There was no case where the interlocks or the valves would have failed to provide isolation. Additionally, each time the plant is cooled down and RHR is placed in shutdown cooling, failures of the valve's interlock system will be detected.

Since these MOVs and interlocks have a good past performance record and the surveillance test can only be performed during refueling conditions, the test interval can be safely extended to 24 months.

In addition to extending the surveillance interval the Authority is proposing the removal of the asterisked note associated with item 13. This note is no longer necessary since the length of the surveillance interval is already identified. This proposed change is consistent with the Westinghouse Standard Technical Specifications.

Residual Heat Removal System Leakage Test

The purpose of this test is to verify the integrity of the section of Residual Heat Removal System lying outside the containment. The test shall be conducted at every refueling in accordance with Technical Specification Section 4.4.I.

This test can only be performed when the Residual Heat Removal System is in service. Visual inspection is performed for excessive leakage: the leakage is not to exceed two gallons per hour from components of the RHR system with the RHR pump running for a minimum of 30 minutes with the pressure of the suction and discharge piping greater than 350 psig.

Since the RHR system cannot be tested during normal plant operation, the system leakage test and the visual inspection can only be performed during residual heat removal system operation and refueling outages.

A review of the RHR system leakage test results from 8/87 through 7/92 indicated that leakage was below the acceptance criterion for each of the four refueling intervals. In the 1990 test, no leakage was detected. In the other two instances, the leakage was attributed to normal seal leakages from valve stems (in 1989), and from one RHR pump (in 1987). In April of 1992 a seal leak was detected in a safety injection pump during the performance of the monthly pump functional test. This leakage was conservatively attributed to the residual heat removal system outside containment and as such slightly exceeded the surveillance acceptance criterion of two gallons per hour.

Unless the plant is in a residual heat removal mode with fuel in the core, the RHR system is not

in operation, and the likelihood of leaks developing while the system is in standby is minimal. Extending the surveillance interval would only extend the period when the system is in standby condition. The system is designed, constructed and maintained to standards which minimize the possibility of developing leaks. The system integrity has been amply demonstrated over the last four test cycles. Since past test data supports the integrity of the system and an extended standby period is not expected to increase leakage, there is a reasonable expectation that the RHR system will operate without excessive leakage.

Based on a good past inspection record, and the low likelihood of increased leakage, the test interval can be safely extended to 24 months.

Recirculation Pump Test

The purpose of this test is to demonstrate the operability of Recirculation Pumps 31 and 32, and check valves 886A and 886B. Test data is obtained and analyzed by the Inservice Inspection Program for assessment of recirculation pumps and check valves operational readiness. The test is in accordance with plant Technical Specification Section 4.5.B.1.b.

IP-3 Technical Specification Section 4.5.B.1.b. requires that the recirculation pumps shall be started during reactor shutdown for refueling. Acceptable levels of performance shall be that the pumps start, reach their required developed head on recirculation flow, and operate for at least 15 minutes. Check valves 886A and B are verified open with their associated pump operating and shut and back seated with operation of the other pump.

During a typical non-refuel outage, the extent of the preparations required and the length of time needed for testing these pumps would significantly affect plant availability, thus, such testing is considered impractical. If this testing interval is not extended with the refueling cycle, performing this test would create approximately 5,000 extra gallons of contaminated waste water which would require processing.

This test should be performed when the plant is in the cold shutdown condition only. Testing these pumps during plant operation is impractical since they are located inside containment and are maintained in a dry condition. In addition, since these pumps stand idle and dry except for periods of testing, significant inservice degradation is unlikely.

A review of the last five years of test results revealed that only once, during the 5/89 outage, recirculation pump #31 would not rotate due to a stuck rotor caused by personnel error. A modification performed prior to the test had improperly connected the motor leads so that the motor rotated in the reverse direction therefore resulting in the stuck rotor. Work requests and a modification replaced the recirculation pump #31 motor shaft, and adjusted 31 & 32 pump packing glands. The work history review on valves 886A and 886B indicates no valve failures over the past five years.

Based on the good surveillance test history and the availability of the redundant RHR external recirculation system, this surveillance test interval can be safely extended to 24 months.

Safety Injection/Residual Heat Removal System Check Valve Operability Tests

Technical Specification Sections 4.5.B.2.b, 4.5.B.2.c and 4.5.B.2.d require demonstrating the operability of the low head injection line/SI accumulator check valves (897A, 897B, 897C, 897D), the RHR return/low head injection check valves (838A, 838B, 838C, 838D) and the safety injection system accumulator check valves (895A, 895B, 895C, 895D) and the operability of the safety injection hi-head check valves (857), respectively. During each refueling outage these valves are part-stroke exercised and a leakage test performed to verify closure. In addition, during each reactor refueling outage, a sample disassembly and inspection on these check valves is performed to verify full-stroke open capability in accordance with NRC generic letter 89-04. If any operation is performed after the initial leak test which could unseat the check valves, the leak test is repeated for the affected valves.

Valves 897A through 897D

Valves 897A through 897D supply make-up from the RHR/low head safety injection pumps or the safety injection accumulators to the RCS cold legs and isolate those components from RCS pressure during normal plant operation. Neither the RHR/low head safety injection pumps nor the safety injection accumulators can provide enough pressure to overcome RCS pressure; thus, exercising these valves during plant operation is not possible.

Valves 838A through 838D

Valves 838A through 838D are the loop isolation valves for RHR return and low head injection. They open for low pressure safety injection (LPSI) and RHR cooling and remain closed for primary system pressure isolation. The only means of verifying valve closure is to perform a leakage test, which is impractical during plant operation or a short-duration outage. At least once every two years these valves are exercised and leakage tests performed to verify closure.

Valves 895A through 895D

Valves 895A through 895D open to provide safety injection flow into the RCS cold legs and close to provide pressure isolation between the RCS and SI accumulators. Exercising these valves to the open position requires actuation of safety injection and overcoming the pressure of RCS. This cannot be done during normal plant operation since the maximum accumulator pressure is considerably less than that of the reactor coolant system.

Valves 895A through D are seldom operated, therefore, valve degradation as a result of wear or abuse is not likely. A partial-stroke test followed by a leak rate test adequately ensures that a valve of this type is intact and functioning properly. Any significant deterioration of the valve internals will be discovered during a leak test.

During each reactor refueling outage, one of valves 895A-D is disassembled, inspected, and manually exercised to verify operability. The schedule is rotated such that all valves are inspected in sequence. During these inspections, should a disassembled valve prove

to be inoperable (i.e. incapable of performing its safety function), then, during the same outage, the remainder of the subject valves will be disassembled, inspected, and exercised to verify operability.

A review of the surveillance test history and operating occurrence reports from 1985 through 1992 revealed that the past valve test results were satisfactory. Work requests were issued mainly for preventive maintenances and check valve disassembly and inspections per ISI Program. During 1989, the retaining block studs on Anchor Darling Check Valves SI-895A,B,C,D and SI-897A,B,C,D were replaced with type A564 stainless steel material in accordance with NRC IE Notice 88-85.

Safety Injection Hi-Head Check Valves (857's)

Safety Injection Hi-Head Check Valves (857's) cannot be exercised during plant operation since the safety injection pumps cannot develop sufficient head to open them against normal operational RCS pressure. During cold shutdown, exercising these valves would require operation of the SIS pumps and injection into the reactor coolant loops. This has the potential of causing low-temperature over-pressurization of the RCS.

During each reactor refueling outage, valves 857A, B, G, H, Q through U, and W will be exercised and each valve individually leakage tested to verify closure in accordance with the Inservice Testing program.

In order to perform a full-stroke test of valves 857 C through F, J through N, and P, the boron injection tank (BIT) must be drained and a refueling outage in progress. Draining this tank represents a considerable effort with the potential of creating major problems related to the propensity of boron crystallization and plate-out in non-heat-traced portions of the drain piping. This evolution also creates a significant amount of contaminated waste.

During each reactor refueling outage, these valves (857 C through F, J through N, and P) are partial-stroke exercised by limited injection flow through the BIT bypass line followed by a leakage test performed to verify closure. Each of these valves will be full-stroke tested whenever the BIT is drained and other plant conditions permit.

In addition, during each reactor refueling outage, at least two of these valves (857 C through F, J through N, and P) are disassembled, inspected, and manually exercised to verify operability. The schedule is rotated such that all valves are inspected in sequence and the interval between inspections for each valve will not exceed six (6) years. During these inspections, should a disassembled valve prove to be inoperable (i.e. incapable of performing its safety function), then, during the same outage, the remainder of the subject valves will be disassembled, inspected, and exercised to verify operability.

A review of the last five years of test results revealed that all test results were satisfactory, and no repair work or work request orders were issued on these Safety Injection Hi-Head check valves (SI-857's).

Since these valves have a good past performance record, and the surveillance test can only be

performed during cold shutdown conditions, the test intervals can be safely extended to 24 months.

Rescission of Confirmatory Order

In order to provide 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, the Authority is proposing a Technical Specification Change to implement the requirements set forth in NRC generic letter dated February 23, 1980, regarding testing of LPI/RHR check valves.

Section III - No Significant Hazards Evaluation

Consistent with the requirements of 10 CFR 50.92, the enclosed application is judged to involve no significant hazards based on the following information:

(1) Does the proposed license amendment involve a significant increase in the probability or consequences of any accident previously evaluated?

Response:

The proposed changes do not involve a significant increase in the probability or consequences of any accident previously analyzed. These changes propose extending the surveillance intervals for safety injection/residual heat removal systems testing. The proposed change to specification 4.5.B.2.d follows the guidance of NRC generic letter dated, February 23, 1980. The change to bases on page 4.1-3 simply includes 24 months in the calibration discussion. The removal of an asterisked note to item 13 of Table 4.1-3 provides consistency with the Westinghouse Standard Technical Specifications. The changes do not involve any physical changes to the plant, nor do they alter the way any equipment functions. An evaluation of past equipment performance and other means of detecting system problems provides assurance that the longer surveillance intervals will not degrade system performance.

(2) Does the proposed license amendment create the possibility of a new or different kind of accident from any previously evaluated?

Response:

The proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated. These changes propose extending the surveillance intervals for safety injection/residual heat removal testing. The proposed change to specification 4.5.B.2.d follows the guidance of NRC generic letter dated, February 23, 1980. The changes to bases on page 4.1-3 simply include 24 months in the calibration discussion. The removal of an asterisked note to item 13 of Table 4.1-3 provides consistency with the Westinghouse Standard Technical Specifications. The changes do not involve any physical

changes to the plant, nor do they alter the way any equipment functions. An evaluation of past equipment performance and other means of detecting system problems provides assurance that the longer surveillance intervals will not degrade system performance.

(3) Does the proposed amendment involve a significant reduction in a margin of safety?

Response:

The proposed changes do not involve a significant reduction in a margin of safety. These changes propose extending the surveillance intervals for safety injection/residual heat removal systems testing. The proposed change to specification 4.5.B.2.d follows the guidance of NRC generic letter dated, February 23, 1980. The changes to bases on page 4.1-3 simply includes 24 months in the calibration discussion. The removal of an asterisked note to item 13 of Table 4.1-3 provides consistency with the Westinghouse Standard Technical Specifications. The changes do not involve any physical changes to the plant, nor do they alter any equipment functions or system setpoints. An evaluation of past equipment performance and other means of detecting system problems provides assurance that the longer surveillance intervals will not degrade system performance.

Section IV - Impact of Changes

These changes will not adversely impact the following:

ALARA Program
Security and Fire Protection Programs
Emergency Plan
FSAR and SER Conclusions
Overall Plant Operations and the Environment

Section V - Conclusions

The incorporation of these changes: 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 Analysis Report; c) will not reduce the margin of safety as defined in the bases for any technical specification; d) does not constitute an unreviewed safety question; and e) involves no significant hazards considerations as defined in 10 CFR 50.92.

Section VI - References

1) IP3 SER
2) IP3 FSAR
3) Indian Point 3 - 24 Month Operating Cycle Safety Injection/Residual Heat Removal Systems Surveillance and Maintenance Extensions, Doc. No: IP3-RPT-SI-00399, Rev. 0, December, 1992.