ATTACHMENT I TO IPN-92-046

PROPOSED TECHNICAL SPECIFICATION CHANGES

RELATED TO

REACTOR COOLANT SYSTEM TESTING

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

TABLE 4.1-1 (Sheet 3 of 5)

Channel Description	Check	Calibrate	Test	Remarks
23. Temperature Sensor in Auxiliary Boiler Feedwater Pump Building	N.A.	N.A.	18M	
24. Temperature Sensors in Primary Auxiliary Building				
a. Piping Penetration Area	N.A.	N.A.	18M	
b. Mini-Containment Area	N.A.	N.A.	18M	
c. Steam Generator Blowdown Heat Exchanger Room	N.A.	N.A.	18M	
25. Level Sensors in Turbine Building	N.A.	N.A.	18M	
26. Volume Control Tank Level	N.A.	18M	N.A.	
27. Boric Acid Makeup Flow Channel	N.A.	18M	N.A.	
28. Auxiliary Feedwater:				
a. Steam Generator Level	S	18M	Q	Low-Low
b. Undervoltage	N.A.	18M	18M	
c. Main Feedwater Pump Trip	N.A.	N.A.	18M	
29. Reactor Coolant System Subcooling Margin Monitor	D	18M	N.A.	
30. PORV Position Indicator	N.A.	N.A.	24M	Limit Switch
31. PORV Position Indicator	D	24M	24M	Acoustic Monitor
32. Safety Valve Position Indicator	D	24M	24M	Acoustic Monitor
33. Auxiliary Feedwater Flow Rate	N.A.	18M	N.A.	

Amendment No. 38, 63, 74, 93, 100, 107, 123,

TABLE 4.1-1 (Sheet 5 of 5)

Channel Description	Check	Calibrate	Test	Remarks
43. Reactor Trip Bypass Breakers	N.A.	N.A.	(1)	1) Manual shunt trip prior to each use
			18M(2)	2) Independent operation of under-voltage and shunt trip from Control Room manual push-button
			18M(3)	3) Automatic undervoltage trip
44. Reactor Vessel Level Indication System (RVLIS)	D	18M	N.A.	
45. Ambient Temperature Sensors Within the Containment Building	D	18M	N.A.	
46. River Water Temperature # (installed)	S	18M	N.A.	1) Check against installed instrumentation or another portable device
47. River Water Temperature # (portable)	S (1)	Q (2)	N.A.	2) Calibrate within 30 days prior to use and quarterly thereafter

* By means of the movable incore detector system

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

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

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

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

Amendment No. 38, 34, 63, 74, 78, 93, 98, 107, 123,

TABLE 4.1-3 (Sheet 1 of 2)

FREQUENCIES FOR EQUIPMENT TESTS		
	Check	Frequency
1. Control Rods	Rod drop times of all control rods	18M
2. Control Rods	Movement of at Least 10 steps in any one direction of all control rods	Every 31 days during reactor critical operations
3. Pressurizer Safety Valves	Set Point	24M
4. Main Steam Safety Valves	Set Point	18M
5. Containment Isolation System	Automatic actuation	18M
6. Refueling System Interlocks	Functioning	Each refueling, prior to movement of core components
7. Primary System Leakage	Evaluate	5 days/week
8. Diesel Generators Nos. 31, 32, & 33 Fuel Supply	Fuel Inventory	Weekly
9. Turbine Steam Stop Control Valves	Closure	Yearly
10. L.P. Steam Dump System (6 lines)	Closure	Monthly
11. Service Water System	Each pump starts and operates for 15 minutes (unless already operating)	Monthly
12. City Water Connections to Charging Pumps and Boric Acid Piping	Temporary connections available and valves operable	18M

Amendment No. 10, 14, 43, 83, 93, 99, 125,

TABLE 4.1-3 (Sheet 2 of 2)

13. RHR Valves 730 and 731	Automatic isolation and interlock action	18M*
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

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

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

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.

Amendment No. 10, 38, 63, 93, 99, 123,

ATTACHMENT II TO IPN-92-046

SAFETY EVALUATION

RELATED TO

REACTOR COOLANT SYSTEM TESTING

TECHNICAL SPECIFICATION CHANGES

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 reactor coolant 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:

- Safety Valve Set Pressure check,
- Safety Valve Position Indicator calibration and testing,
- PORV and PORV Block Valve operability tests,
- PORV Position Indicator calibration and testing (limit switch and acoustic monitor), and
- Reactor Vessel Head Vent operability checks.

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 an 18 month surveillance outage or an extended mid-cycle outage, changes are required to the reactor coolant 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. Starting below is an evaluation for each technical specification that this application proposes to change.

Safety Valve Set Pressure Check

The purpose of this test is to verify the operational readiness of the pressurizer safety valves. According to ANSI/ASME OM-1-1981, a minimum of two safety valves have to be tested within a 24 month period, and all safety valves have to be tested at least once in a five year period. Indian Point 3 Technical Specifications require all three safety valves to be tested every refueling.

A review of the surveillance test results and occurrence reports for the past five years, showed that safety valve performance has been good, except for actuation within one percent of the set pressure. As specified in the IP3 Technical Specifications, the safety valve lift settings shall be 2485 psig with a $\pm$ 1% margin. During the 1989 and 1990 surveillance tests, two valves exceeded the 1% margin (the largest deviation was 2% of the setpoint).

Safety valve set pressure check test results from 1987 to 1991 were evaluated for evidence of increasing setpoint drift with a longer length of service. There does not seem to be any trend of increasing drift with an increasing length of service. Therefore, it is believed that the IP3 safety valve setpoint drift is not time dependent and will not be affected by extending the surveillance interval to once every 24 months. Also, the acoustic flow monitor alarms provide assurance that safety valve malfunctions (such as partially open valves) occurring during normal plant operation

will be detected.

Since the safety valves are being tested at a frequency greater than that required by the ASME Code, and partially open safety valves are detected and alarmed by temperature elements or acoustic flow monitors during normal plant operation, the pressurizer safety valve set pressure and seat leakage test can safely be extended with the longer operating cycle.

Safety Valve Position Indicator Calibration and Testing

The acoustic monitoring system provides the control room with positive indication that a safety valve is open. The valve flow monitor module processes the voltage signal by flow noise and pipe vibration at the valve tailpiece and indicates the relative value of flow on a lighted bar graph display calibrated in 10% increments of full flow. If any monitor channel detects a flow signal greater than 25% of the full flow signal, it will trigger a common pressurizer PORV and safety valve acoustic monitoring alarm in the control room.

A review of calibration results between 1987 and 1990 indicates that the output voltages were slightly out of specification at the low end (flow indications between 0.01 and 0.04) of the instrument span. An instrument drift evaluation of the calibration data indicates that the as-found condition of the acoustic monitors are consistently within the calibration tolerance (5% of calibration span) at 25% flow. Since a flow indication signal greater than 0.25 will trigger the acoustic monitor alarm, any increased drift at the low end of the instrument span should not affect the operability of the acoustic monitoring alarm. Additionally, a best estimate of drift over 30 months was determined to be about 1.4% of the calibration span, well within the calibration tolerance. The best estimate of drift was determined as follows: (1) for each of the three safety valves, worst case 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.

PORV and PORV Block Valve Operability Tests

The PORVs and PORV block valves are stroke tested to demonstrate that they are operable. These valves are tested while the plant is shutdown, due to the potential adverse affects (pressure transients) of a PORV opening prematurely or failing to close.

The PORV and PORV block valve past surveillance test results (from 1986 to 1991) were all satisfactory. However, PORV leaks have occurred during operation. These leaks were detected, and isolated by the PORV block valve.

Operability of the PORV during power operation is not critical for overpressure protection, because the pressurizer code safety valves protect the Reactor Coolant System from overpressurization, without the aid of either PORV. If any PORV is inoperable during power operation, the block valve is closed to prevent an inoperable PORV from inadvertently opening. The Technical Specification for the operability check of the block valves can be increased from each refueling outage to quarterly (unless the block valve is shut due to a leaking/inoperable PORV), to be consistent with the ASME section XI program.

The PORV test interval can be safely extended with the longer outage because a review of PORV stroke tests for the past five years show that the refueling test is not being relied on to detect PORV related deficiencies, the PORV test can not be performed during normal plant operation, and malfunctions (such as partially open valves) occurring during normal operation will be detected by the temperature elements or acoustic flow monitors, which are indicated (and have associated alarms) in the control room. Additionally, the specification for frequency of block valve operability checks is being increased to quarterly.

PORV Position Indicator Calibration and Testing

The acoustic monitoring system provides the control room with positive indication that a safety valve is open. The valve flow monitor module processes the voltage signal by flow noise and pipe vibration at the valve tailpiece and indicates the relative value of flow on a lighted bar graph display calibrated in 10% increments of full flow. If any monitor channel detects a flow signal greater than 25% of the full flow signal, it will trigger a common pressurizer PORV and safety valve acoustic monitoring alarm in the control room.

A review of calibration results between 1987 and 1990 indicates that the output voltages were slightly out of specification at the low end (flow indications between 0.01 and 0.04) of the instrument span. An instrument drift evaluation of the calibration data indicates that the as-found condition of the acoustic monitors are consistently within the calibration tolerance (5% of calibration span) at 25% flow. Since a flow indication signal greater than 0.25 will trigger the acoustic monitor alarm, any increased drift at the low end of the instrument span should not affect the operability of the acoustic monitoring alarm. Additionally, a best estimate of drift over 30 months was determined to be about 1.4% of the calibration span, well within the calibration tolerance.

The PORV limit switch functional test is being extended to once per 24 months consistent with the PORV stroke test. The calibration of the limit switches is not critical for PORV position indication and is only performed when the PORV stroke test shows switch adjustment is required, not every refueling cycle. Therefore, the Technical Specification for refueling interval limit switch calibration is being deleted. This clarifies the specifications to be consistent with past experience, since actual limit switch adjustment is more appropriately considered a maintenance activity.

Limit switch testing can be safely extended, and calibration deleted, because: the acoustic monitoring system provides positive indication of PORV position, partially open valves will be detected and alarmed, and the stroke time testing of the PORVs provides assurance that the limit switches are functional.

Reactor Vessel Head Vent Operability Checks

The reactor vessel head vent system is provided for the removal of noncondensable gases or steam from the reactor vessel head. The system is designed to mitigate a possible condition of inadequate core cooling due to loss of natural circulation resulting from the accumulation of noncondensable gases in the reactor coolant system. The reactor vessel head vent valves are normally closed and de-energized during the entire operating cycle, except briefly during startup and shutdown, to prevent inadvertent operation that could result in a small break loss-of-coolant-

accident inside containment. These valves can only be tested while the reactor is in the cold shutdown condition.

A review of past stroke tests from 1986 to 1991 showed no valve failures. However, failure of the indicating lights has occurred. For example, in March 1990, the indicator lights for valves SOV-652 and 654 did not operate properly. Because these valves have demonstrated good stroke test performance in the past, the surveillance interval can be safely extended to once per 24 months.

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 reactor coolant systems testing. 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 reactor coolant systems testing. 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 reactor coolant systems testing. 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 this change: 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