ATTACHMENT I TO IPN-92-009

PROPOSED TECHNICAL SPECIFICATION CHANGES

RELATED TO

CONTROL ROD DRIVE TESTING

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

TABLE 4.1-1 (Sheet 1 of 5)

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTS OF INSTRUMENT CHANNELS

Channel Description	Check	Calibrate	Test	Remarks
1. Nuclear Power Range	S	D (1) M (3)*	Q (2)** Q (4)	1) Heat balance calibration 2) Bistable action (permissive, rod stop, trips) 3) Upper and lower chambers for axial offset 4) Signal to Δ T
2. Nuclear Intermediate Range	S (1)	N.A.	P (2)	1) Once/shift when in service 2) Verification of channel response to simulated inputs
3. Nuclear Source Range	S (1)	N.A.	P (2)	1) Once/shift when in service 2) Verification of channel response to simulated inputs
4. Reactor Coolant Temperature	S	R	Q (1) Q (2)	1) Overtemperature - Δ T 2) Overpower - Δ T
5. Reactor Coolant Flow	S	R	Q	
6. Pressurizer Water Level	S	R	Q	
7. Pressurizer Pressure	S	R	Q	High and Low
8. 6.9 KV Voltage & Frequency	N.A.	R	Q	Reactor protection circuits only
9. Analog Rod Position	S	24M	M	

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

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
			R(2)	2) Independent operation of under-voltage and shunt trip from Control Room manual push-button
			R(3)	3) Automatic undervoltage trip
44. Reactor Vessel Level Indication System (RVLIS)	D	R	N.A.	
45. Ambient Temperature Sensors Within the Containment Building	D	R	N.A.	
46. River Water Temperature # (installed)	S	R	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
R - Each refueling outage
TM - At least every two months on a staggered test basis (i.e., one train per month)
24M - At least once per 24 months

Amendment No. 38, 54, 63, 74, 78, 93, 98, 107,

TABLE 4.1-3 (Sheet 1 of 2)

FREQUENCIES FOR EQUIPMENT TESTS		
	<u>Check</u>	<u>Frequency</u>
1. Control Rods	Rod drop times of all control rods	24M
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	R
4. Main Steam Safety Valves	Set Point	R
5. Containment Isolation System	Automatic actuation	R
6. Refueling System Interlocks	Functioning	R (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	R

TABLE 4.1-3 (Sheet 2 of 2)

13. RHR Valves 730 and 731	Automatic isolation and interlock action	R*
14. PORV Block Valves	Operability through 1 complete cycle of full travel	R
15. PORV Valves	Operability	R
16. Reactor Vessel Head Vents	Operability	R

R Each Refueling Outage
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.

ATTACHMENT II TO IPN-92-009

SAFETY EVALUATION

RELATED TO

CONTROL ROD DRIVE TESTING

TECHNICAL SPECIFICATION CHANGES

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
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Section I - Description of Changes

This application for amendment to the Indian Point 3 Technical Specifications proposes to change the frequency of control rod drop time testing (specified in Table 4.1-3) and calibration of analog rod position (specified in Table 4.1-1) to accommodate operation with a 24 month operating cycle. Starting with cycle nine (scheduled to start in June, 1992), Indian Point 3 will begin 24 month operating cycles, instead of the current 18 month cycles.

Section II - Evaluation of Changes

Once each refueling outage (currently about every 18 months) control rod drop times are verified to ensure that the drop times are consistent with the drop times assumed in the plant's safety analysis. Also, rod position indicators are calibrated each refueling outage to ensure that the positions of individual control rods can be determined accurately, and that misalignment of an individual rod will be detected. Performing these tests at power would result in a reactor trip and subsequent challenges to safety systems. Therefore, in order to accommodate 24 month operating cycles, these surveillances need to be extended (to be performed once every 24 months).

Control Rod Drop Test

During start-up from each refueling outage, each control rod is drop time tested at operating temperature and full flow. The drop time, measured from loss of stationary gripper coil voltage to dashpot entry, is shown to be no greater than 2.4 seconds.

Drop time testing is performed after fuel handling and reactor head reassembly have been completed. This ensures that the refueling activity itself has not affected rod drop time (for example, by increased friction from changes in component clearances or alignments).

The longer operating cycle will increase the time between post-refueling rod drop time tests. The fuel assembly design ensures that changes in component clearances or physical configuration, which could potentially affect rod drop times, will not occur. Additionally, control rod operability is checked once every 31 days by moving the rods at least 10 steps in any one direction.

Rod Position Indicator Calibrations

The rod position indicator (RPI) system is calibrated during start-up following each refueling outage with the reactor in the hot shutdown condition. RPI operability is continuously ensured by channel checks each shift, functional tests of the RPI rod bottom bistables each month, and observation of core instrumentation. Additionally, control room annunciators and alarms are available, as necessary, to indicate abnormal conditions. The on-line surveillances supplement the once per cycle calibration, and provide adequate assurance that rod misalignment will be detected.

The longer operating cycle will increase the time between the refueling calibrations. Postulated increases in instrument drift associated with the longer time interval are expected to remain within system accuracies. If excessive drift were to occur, it would be detected quickly by one or more of the RPIs not in agreement with the pulse counter (demand position). This condition would be indicated prior to exceeding core safety limits.

The Authority reviewed the historical performance of the control rod drive system equipment. An evaluation of the surveillance testing and maintenance history of the control rod drive system determined that the past performance of the system has been reliable. A review of significant occurrence reports from 1985 through mid 1991 indicates that equipment problems are being identified and corrected without relying on the once per refueling outage tests to identify performance problems.

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 an accident previously evaluated?

Response:

The proposed changes do not involve a significant increase in the probability or consequences of a previously-analyzed accident. The changes propose extending the surveillance intervals for rod drop time testing and rod position indication calibration. The changes do not involve any physical changes to the plant, nor do they alter the way any equipment functions. On-line surveillances and testing assure equipment operability, and also assure that any control rod misalignment will be detected. A review of significant occurrence reports from 1985 through mid 1991 indicates that equipment problems are being identified and corrected without relying on the once per refueling outage tests to identify performance problems.

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

Response:

The proposed license amendment does not create the possibility of a new or different kind of accident. The changes propose extending the surveillance intervals for rod drop time testing and rod position indication calibration. The changes do not involve any physical changes to the plant, nor do they alter the way any equipment functions. On-line surveillances and testing assure equipment operability, and also assure that any control rod misalignment will be detected.

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

Response:

The proposed amendment does not involve a significant reduction in a margin of safety. The changes propose extending the surveillance intervals for rod drop time testing and rod position indication calibration. The changes do not involve any physical changes to the plant, nor do they alter the way any equipment functions. On-line surveillances and testing assure equipment operability, and also assure that any control rod misalignment will be detected. A review of significant occurrence reports from 1985 through mid 1991 indicates that equipment problems are being identified and corrected without relying on the once per refueling outage tests to identify performance problems.

Section IV - Impact of Changes

These changes will not adversely impact the following:

ALARA Program
Security and Fire Protection Programs
Emergency Plan
FSAR or 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

a) IP-3 FSAR
b) IP-3 SER