ATTACHMENT I TO IPN-89-071

PROPOSED TECHNICAL SPECIFICATION CHANGES

RELATED TO

REACTOR PROTECTION SYSTEM AND ENGINEERED SAFETY FEATURES

ACTUATION SYSTEM SURVEILLANCE INTERVALS

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

Almost all reactor protection channels are supplied with sufficient redundancy to provide the capability for channel calibration and test at power. Exceptions are backup channels such as reactor coolant pump breakers. The removal of one trip channel on process control equipment is accomplished by placing that channel bistable in a tripped mode; e.g., a two-out-of-three circuit becomes a one-out-of-two circuit. A channel bistable may also be placed in a bypassed mode; e.g., a two-out-of-three circuit becomes a two-out-of-two circuit. The nuclear instrumentation system channels are not intentionally placed in a tripped mode since the test signal is superimposed on the normal detector signal to test at power. Testing of the NIS power range channel requires: (a) bypassing the Dropped Rod protection from NIS, for the channel being tested; and (b) defeating the ΔT protection CHANNEL SET that is being fed from the NIS channel and (c) defeating the power mismatch section of T_{avg} control channels when the appropriate NIS channel is being tested. However, the Rod Position System and remaining NIS channels still provide the dropped-rod protection. Testing does not trip the system unless a trip condition exists in a concurrent channel.

In the event that either the specified Minimum Number of Operable Channels or the Minimum Degree of Redundancy cannot be met, the reactor and the remainder of the plant is placed, utilizing normal operating procedures, in that condition consistent with the loss of protection.

The source range and the intermediate range nuclear instrumentation and the turbine and steam-feedwater flow mismatch trip functions are not required to be operable since they were not used in the transient and safety analysis (FSAR Section 14).

<u>References:</u>

1) FSAR - Section 7.5

2) FSAR - Section 14.3

3) FSAR - Section 14.2.5

2.5 x 10^{-6} failure/hrs. per channel. This is based on operating experience at conventional and nuclear plants. An unsafe failure is defined as one which negates channel operability and which, due to its nature, is revealed only when the channel is tested or attempts to respond to a bona fide signal.

For a specified test interval W and an M out of N redundant system with identical and independent channels having a constant failure rate λ, the average availability A is given by:

$$A = \frac{W - Q\left(\frac{W}{N-M+2}\right)}{W} = 1 - \frac{N!}{(N-M+2)!\,(M-1)!}(\lambda W)^{N-M+1}$$

where A is defined as the fraction of time during which the system is functional, and Q is the probability of failure of such a system during a time interval W.

For a 2-out-of-3 system A = 0.9999708, assuming a channel failure rate, λ, equal to 2.5 x 10^{-6} hr^{-1} and a test interval, W, equal to 2160 hrs.

This average availability of the 2-out-of-3 system is high, hence the test interval of one quarter is acceptable.

Because of their greater degree of redundancy, the 1/3 and 2/4 logic arrays provide an even greater measure of protection and are thereby acceptable for the same testing interval. Those items specified for quarterly testing are associated with process components where other means of verification provide additional assurance that the channel is operable, thereby requiring less frequent testing.

Specified surveillance intervals for the Reactor Protection System and Engineered Safety Features have been determined in accordance with WCAP - 10271, Supplement 1, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," and WCAP - 10271, Supplement 2, Revision 1, "Evaluation of Surveillance Frequencies and Out of Service Times for the Engineered Safety Features Actuation System," as approved by the NRC and documented in the SER'S (letters to J. J. Sheppard from C. O. Thomas, dated February 21, 1985, and to R. A. Newton from C. W. Rossi, dated February 22, 1989). Surveillance intervals were determined based on maintaining an appropriate level of reliability of the Reactor Protection System and Engineered Safety Features instrumentation.

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 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 - T 2) Overpower - 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	R	M	

TABLE 4.1-1 (Sheet 2 of 5)

Channel Description	Check	Calibrate	Test	Remarks
10. Steam Generator Level	S	R	Q	
11. Residual Heat Removal Pump Flow	N.A.	R	N.A.	
12. Boric Acid Tank Level	S	R	N.A.	Bubbler tube rodded during calibration
13. Refueling Water Storage Tank Level	W	R	N.A.	Low level alarms
14. Containment Pressure	S	R	Q	High and High-High
15. Process and Area Radiation Monitoring Systems	D	R	Q	
16. Containment Water Level Monitoring System:				
a. Containment Sump	N.A.	R	N.A.	Narrow Range, Analog
b. Recirculation Sump	N.A.	R	N.A.	Narrow Range, Analog
c. Containment Water Level	N.A.	R	N.A.	Wide Range
17. Accumulator Level and Pressure	S***	R	N.A.	
18. Steam Line Pressure	S	R	Q	
19. Turbine First Stage Pressure	S	R	Q	
20. Reactor Protection Relay Logic	N.A.	N.A.	TM	
21. Turbine Trip				
a. Independent Overspeed	N.A.	R	M	Electrical
b. Low Auto Stop Oil Pressure	N.A.	R	N.A.	
22. Boron Injection Tank Return Flow	S	R	N.A.	

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.	R	
24. Temperature Sensors in Penetration Area of Primary Auxiliary Building	N.A.	N.A.	R	
25. Level Sensors in Turbine Building	N.A.	N.A.	R	
26. Volume Control Tank Level	N.A.	R	N.A.	
27. Boric Acid Makeup Flow Channel	N.A.	R	N.A.	
28. Auxiliary Feedwater:				
a. Steam Generator Level	S	R	Q	Low-Low
b. Undervoltage	N.A	R	R	
c. Main Feedwater Pump Trip	N.A	N.A.	R	
29. Reactor Coolant System Subcooling Margin Monitor	D	R	N.A.	
30. PORV Position Indicator	N.A.	R	R	Limit Switch
31. PORV Position Indicator	D	R	R	Acoustic Monitor
32. Safety Valve Position Indicator	D	R	R	Acoustic Monitor
33. Auxiliary Feedwater Flow Rate	N.A.	R.	N.A.	

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TABLE 4.1-1 (Sheet 4 of 5)

Channel Description	Check	Calibrate	Test	Remarks
34. Plant Effluent Radioiodine/ Particulate Sampling	N.A.	N.A.	R	Sample line common with monitor R-13
35. Loss of Power				
a. 480v Bus Undervoltage Relay	N.A.	R	M	
b. 480v Bus Degraded Voltage Relay	N.A.	R	M	
c. 480v Safeguards Bus Undervoltage Alarm	N.A.	R	M	
36. Main Steam Line Radiation Monitors	D	R	Q	R-62A, B,C,D
37. Containment Hydrogen Monitors	D	Q	M	
38. Wide Range Plant Vent Monitor	D	R	Q	R-27
39. High Range Containment Radiation Monitors	D	R	Q	R-25, R-26
40. Core Exit Thermocouples	D	N.A.	N.A.	
41. Overpressure Protection System (OPS)	D	R	R	
42. Reactor Trip Breakers	N.A.	N.A.	TM(1)	1) Independent operation of undervoltage and shunt trip attachments
			R(2)	2) Independent operation of undervoltage and shunt trip from Control Room manual push-button

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 undervoltage 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.	

* 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.

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)

<u>ATTACHMENT II TO IPN-89-071</u>

<u>SAFETY EVALUATION</u>

<u>RELATED TO</u>

<u>REACTOR PROTECTION SYSTEM AND ENGINEERED SAFETY</u>

<u>FEATURES ACTUATION SYSTEM SURVEILLANCE INTERVALS</u>

<u>TECHNICAL SPECIFICATION CHANGES</u>

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Section I - Description of Changes

These proposed changes to the Indian Point 3 Technical Specifications seek to increase the surveillance intervals for the Reactor Protection System (RPS) and Engineered Safety Features (ESF) analog channel operational tests from monthly to quarterly, and allow routine analog channel testing in a bypassed condition instead of a tripped condition. Also sought is a correction to a typographical error introduced through a past amendment request. A letter "N" was mistakenly typed instead of the letter "M" as the last letter in the denominator of the equation on page 4.1-3. In addition, Table 4.1-1, sheet 4 of 5, has been included in this submittal for completeness. The changes would revise the analog channel tests for the following channels:

1. Nuclear Power Range
2. Reactor Coolant Temperature
3. Reactor Coolant Flow
4. Pressurizer Water Level
5. Pressurizer Pressure (High and Low)
6. 6.9 KV Voltage and Frequency
7. Steam Generator Level
8. Containment Pressure
9. Steam Line Pressure
10. Auxiliary Feedwater, Steam Generator Level
11. Turbine First Stage Pressure

Section II - Evaluation of Changes

These changes are among those proposed by the Westinghouse Owners Group (WOG) in WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," Supplement 1 to WCAP-10271, and Supplement 2, Revision 1 to WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Engineered Safety Features Actuation System". As approved by the NRC's Safety Evaluation Reports (SERs) of February 21, 1985 and February 22, 1989, these reports document the basis for the proposed revisions to the Technical Specifications. WCAP-10271, Supplement 1, and Supplement 2, Revision 1, gave consideration on a generic basis to such areas as safety, equipment requirements, human factors and operational impact. It consisted of deterministic and numerical evaluation of the effects of increased test and maintenance times, less frequent surveillance, and testing in bypass.

The following are the responses to conditions imposed by the NRC to allow the use of WCAP - 10271 and it's supplements in amending the Technical Specifications.

1. The first condition is that the staff's acceptance of less frequent surveillance for analog channels is contingent on implementation of procedures to identify common cause failures and to test the other channels which may be affected by the common cause.

   Response

   Applicable surveillance procedures will be modified, prior to the institution of quarterly testing, to require an evaluation for common cause failure should any RPS or ESF channel fail during it's quarterly test. Additional testing of other channels in the function would be required if a determination is made that a plausible common cause exists.

2. The second condition is that the staff's acceptance of less frequent surveillance is contingent on confirmation that the instrument setpoint methodology includes sufficient adjustments to offset the drift anticipated as a result of less frequent surveillance.

   Response

   The "as found" and "as left" test data, for a representative sample (i.e. two channels per function) of RPS and ESF analog channels, for twelve months, were examined. The data were grouped into approximately three - month periods during which no adjustments to the bistables were made, such that the data actually involved three months of instrument drift. The data review shows that setpoint drift is random in nature, occurring in both the increasing and decreasing directions. The data review indicates that, for quarterly surveillance testing, the setpoint drift is expected to be bounded by the bistable setpoint tolerances specified in the analog channel test procedures.

3. The third condition requires installed hardware capability for testing in the bypassed mode.

   Response

   Currently, IP3 does not have the hardware capability for full bypass testing. It is NYPA's intention to make hardware changes in the future to provide this testing capability. Once the plant modifications are made, routine testing of the channels with bypass hardware will be performed with the channel in bypass; only those instruments whose hardware capability does not require the lifting of leads or installing of jumpers will be routinely tested in bypass.

4. The fourth condition requires confirmation of the applicability of the generic WCAP - 10271 analyses to the Indian Point 3 (IP3) Nuclear Power Plant.

   Response

   The Westinghouse analyses cover two, three, and four loop plants with either relay or solid state logic. Indian Point 3 is a four loop plant utilizing relay logic. The changes proposed by WCAP - 10271 and its supplements include (1) changing the surveillance frequency of RPS and ESF analog channels from monthly to quarterly, (2) increasing the time an inoperable analog channel may be maintained in an untripped condition from one hour to six hours, (3) increasing the time an inoperable analog channel may be bypassed to allow testing of another channel in the same function from two hours to four hours and allowing the channel test to be done in bypass leaving the inoperable channel tripped, and (4) allowing the testing of analog channels in a bypassed condition instead of a tripped condition. The Westinghouse study and proposed Technical Specification changes are based on a plant with Standard Technical Specifications and installed bypass capability. Indian Point 3 does not have Standard Technical Specifications nor the hardware capability for full bypass testing; when this capability is acquired, routine analog channel testing will be performed in the bypass condition. IP3 is not requesting an increase (to six hours) in the time an inoperable channel may remain untripped. Current plant "Off Normal Operating Procedures" require that the bistable trip switches for the affected instrument channel be placed in the tripped position as one of the operator's "subsequent actions" for an instrument failure. This is a more conservative situation from a safety system availability standpoint than allowing the inoperable channel to remain untripped for up to six hours. The Westinghouse analog channel fault tree model was constructed assuming testing of more than one analog channel at a time. According to the Technical Specifications (section 3.5.2), no more than one channel is tested at a time. This assumption constitutes a more limiting configuration than actual practice from a safety system availability standpoint.

There is one channel, Turbine First Stage Pressure, which appears in this submittal but does not appear in the marked-up Tech. Spec. pages accompanying the WCAP. It is not in the WCAP because it is not a part of Standard Technical Specifications. It is, however, part of the IP3 Tech. Spec. surveillance table (Table 4.1-1). Discussions between Power Authority and Westinghouse personnel have led to the determination that the WCAP analysis is applicable to Turbine First Stage Pressure; this channel is explicitly modeled in the development of the Steam Flow signal in WCAP-10271, Supplement 2, Revision 1 (SPSENS-0013 on Figure B3, page C-105). It is, therefore, appropriate to extend the justification and analysis of the WCAP to include Turbine First Stage Pressure as a channel for which analog channel testing may be extended from monthly to quarterly.

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

   Operation of Indian Point 3 in accordance with the proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   Implementation of the proposed changes is expected to result in an acceptable increase in total Reactor Protection System and Engineered Safety Features yearly unavailability. This increase, which is primarily due to less frequent surveillance, results in an increase of similar magnitude in the probability of an Anticipated Transient Without Scram (ATWS) and in the probability of core melt resulting from an ATWS. The increase also results in a small increase in Core Damage Frequency (CDF) due to unavailability of the ESF signals.

   Implementation of the proposed changes for the RPS is expected to result in a significant reduction in the probability of core melt from inadvertent reactor trips. This is a result of a reduction in the number of inadvertent reactor trips (0.5 fewer inadvertent reactor trips per unit per year) occurring during testing of RPS instrumentation. This reduction is primarily attributable to testing in bypass and less frequent surveillance testing. This reduction of inadvertent core melt probability is sufficiently large to counter the increase in ATWS core melt probability resulting in an overall reduction in total core melt probability. The changes in RPS unavailability and core damage frequency (CDF) and risk are small compared to the error in probabilistic estimates.

The proposed changes do not result in an increase in the severity or consequences of an accident previously evaluated. Implementation of the proposed changes affects the probability of failure of the RPS and ESF, but does not alter the manner in which protection is afforded nor the manner in which limiting criteria are established.

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

<u>Response</u>

The proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes do not result in a change in the manner in which the RPS and ESF provide plant protection. No change is being made which alters the functioning of the RPS or ESF (other than in a test mode). Rather, the likelihood or probability of the RPS or ESF functioning properly is affected as described above. Therefore, the proposed changes do not create the possibility of a new or different kind of accident nor involve a reduction in the margin of safety as defined in the Safety Analysis Report.

The proposed changes do not involve hardware changes, except those necessary to implement testing in bypass. Some existing instrumentation is designed to be tested in bypass, and the current Tech. Specs. do not prohibit testing in bypass. Testing in bypass is also recognized by IEEE Standards. Therefore, testing in bypass does not create the possibility of a new or different kind of accident from any previously evaluated. The proposed changes do not alter the functioning of the RPS or ESF, and so the possibility of a new or different kind of accident from any previously evaluated has not been created.

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

<u>Response</u>

The proposed license amendment does not involve a significant reduction in a margin of safety.

The proposed changes do not alter the manner in which safety limits, limiting safety system setpoints or limiting conditions for operation are determined. The impact of reduced testing, other than as addressed above, is to allow a longer time interval over which instrument uncertainties (e.g., drift) may act. Experience at two Westinghouse plants with extended surveillance intervals has shown the initial uncertainty assumptions to be valid for reduced testing.

Implementation of the proposed changes is expected to result in an overall improvement in safety by :

a. 0.5 fewer inadvertent reactor trips per unit per year. This is due to less frequent testing and testing in bypass, which minimizes the time spent in a partial trip condition.

b. Improvements in the effectiveness of the operating staff in monitoring and controlling plant operation. This is due to less frequent distraction of the operator and shift supervisor to attend to instrument testing.

In the April 6, 1983 Federal Register, Vol. 48, No. 67, pg. 14870, the NRC published a list of examples of amendments that are not likely to involve a significant hazards concern. Example (vi) of that list applies to the proposed changes in the Reactor Protection System and Engineered Safety Features analog channel surveillance intervals and states:

> (vi) A change which either may result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan: for example, a change resulting from the application of a small refinement of a previously used calculational model or design method.

As previously stated, implementation of the proposed changes results in an acceptable increase in the probability of ATWS and ATWS core melt. The probability of core melt due to inadvertent reactor trips is expected to decrease significantly. An overall reduction in total core melt probability is expected. Implementation of the proposed changes does not increase the consequences of a previously analyzed accident nor reduce a margin of safety. Functioning of the RPS and ESFAS and the manner in which limiting criteria are established is unaffected. The stated example of a change which is likely not to involve a significant hazards consideration is applicable, therefore, to the proposed changes.

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 foregoing analysis demonstrates that the proposed amendment to the Indian Point 3 Technical Specifications does not involve a significant increase in the probability or consequences of a previously evaluated accident, does not create the possibility of a new or different kind of accident and does not involve a significant reduction in a margin of safety. Additionally, fewer inadvertent reactor trips are expected, and operator effectiveness is expected to improve. Based upon the preceding analysis, the Authority concludes that the proposed amendment does not involve a significant hazards consideration.

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

a) IP-3 FSAR

b) IP-3 SER

c) WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," January 1983.

d) WCAP-10271, Supplement 1, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," July 1983.

e) Letter from Mr. C. O. Thomas (NRC) to Mr. J. J. Sheppard (WOG-CP&L), dated February 21, 1985 (NRC Safety Evaluation for WCAP-10271).

f) WCAP-10271, Supplement 2, Revision 1, "Evaluation of Surveillance Frequencies and Out of Service Times for the Engineered Safety Features Actuation System," March 1987.

g) Letter from Mr. C. E. Rossi (NRC) to Mr. R. A. Newton (WOG-WEPC), dated February 22, 1989 (NRC Safety Evaluation for WCAP-10271, Supplement 2, Revision 1).