ATTACHMENT I TO IPN-87-011
PROPOSED TECHNICAL SPECIFICATION CHANGES
RELATED TO
REACTOR VESSEL LEVEL INDICATION SYSTEM (RVLIS)

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

TABLE 3.5-5 (Sheet 2 of 3)

PARAMETER	1 NO. OF CHANNELS AVAILABLE	2 MIN. NO. OF CHANNELS REQUIRED**	3 INDICATOR/ RECORDER**
15) Level Sensors in Lower Level of Turbine Building	2	1	ALARM
16) Reactor Coolant System Subcooling Margin Monitor	1	1	RECORDER
17) PORV Position Indicator (Acoustic Monitor)	1/Valve	1/Valve	INDICATOR
18) PORV Position Indicator (Limit Switch)	1/Valve	1/Valve****	INDICATOR AND ALARM
19) PORV Block Valve Position Indicator (Limit Switch)	1/Valve***	1/Valve	INDICATOR
20) Safety Valve Position Indicator (Acoustic Monitor)	1/Valve	1/Valve	INDICATOR
21) Auxiliary Feedwater Flow Rate	1/Pump	1/Pump	INDICATOR
22) Containment Water Level (Wide Range)	2	1	INDICATOR/ RECORDER
23) Containment Hydrogen Monitor	2	1	INDICATOR/ RECORDER
24) High-Range Containment***** Radiation Monitors (R25 R26)	2	1	ALARM
25) Core Exit Thermocouples	4/quadrant	2/quadrant	INDICATOR
26) Reactor Vessel Level Indication System (RVLIS)	2	1	INDICATOR

* One level channel per steam generator (either wide range or narrow range) with at least two wide range channels.

** Columns 2 and 3 may be modified to allow the instrument channel(s) to be inoperable for up to 7 days and/or the recorder to be inoperable for up to 14 days.

*** Except at times when valve operator control circuit is de-energized.

**** Except when the respective block valve is closed.

TABLE 3.5-5 (Sheet 3 of 3)

***** If the high-range containment radiation monitor is determined to be inoperable when the reactor is above the cold shutdown condition, then restore the monitoring capability within 7 days, and

a) Initiate an alternate monitoring capability as soon as practical, but no later than 72 hours after identification of the failure of the monitor. If the monitor is not restored to operable status within 7 days, then,

b) Submit a Special Report to the NRC pursuant to Technical Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system.

\+ If both narrow range analog monitor channels are determined to be inoperable, at least one channel will be restored to operable status within 30 days or the Plant will be brought to hot shutdown within the next 12 hours.

With the exception of the High Range Containment Radiation Monitors, if the minimum number of channels required are not restored to meet the above requirements within the time periods specified, then:

1. If the reactor is critical, it shall be brought to the hot shutdown condition utilizing normal operating procedures. The shutdown shall start no later than at the end of the specified time period.

2. If the reactor is subcritical, the reactor coolant system temperature and pressure shall not be increased more than 25°F and 100 psi, respectively, over existing values.

3. In either case, if the requirements of Columns 2 and 3 are not satisfied within an additional 48 hours, the reactor shall be brought to the cold shutdown condition utilizing normal operating procedures. The shutdown shall start no later than the end of the 48 hours period.

Table 4.1-1 (Sheet 5 of 5)

	Channel Description	Check	Calibrate	Test	Remarks
39.	High Range Containment Radiation Monitoring (R25, R26)	D	R	Q	
40.	Core Exit Thermocouples	D	N.A.	N.A.	
41.	Overpressure Protection System (OPS)	D	R	R	
42.	Reactor Vessel Level Indication System (RVLIS)	D	R	N.A.	

* To be effective after completion of all required modifications.

S -Each Shift
P -Prior to each startup if not done previous week
Q -Quarterly
NA -Not applicable
D -Daily
W -Weekly
M -Monthly
R -Each Refueling Outage

ATTACHMENT II TO IPN-87-011
SAFETY EVALUATION OF PROPOSED
TECHNICAL SPECIFICATIONS RELATED TO
REACTOR VESSEL LEVEL INDICATION SYSTEM (RVLIS)

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
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Section I - Description of Changes

The proposed changes to the Indian Point 3 Technical Specifications are enclosed in Attachment I. The changes relate to the Reactor Vessel Level Indication System (RVLIS). The guidelines and recommendations of Generic Letter No. 83-37 have been utilized in preparing revisions to Tables 3.5-5 and 4.1-1. Editorial changes have also been included in the revised tables (i.e., columns 1, 2 and 3 headings added to Table 3.5-5, Sheet 2 of 3; footnote marked ***** was moved to top of Sheet 3 of 3, Table 3.5-5).

Section II - Evaluation of Changes

The purpose of these proposed changes is to incorporate the appropriate limiting conditions for operation and the surveillance requirements for RVLIS. The installation of RVLIS will be implemented in accordance with the requirements of NUREG-0737, Item II.F.2, "Instrumentation for Detection of Inadequate Core Cooling." RVLIS outputs are displayed on the plant Qualified Safety Parameter Display System. The system performs an input autocalibration sequence by automatically injecting test signals directly into every input on a regular schedule while the system is on line. The proposed limiting conditions of operation (LCO) and surveillance requirements are consistent with other Post Accident Monitoring Systems LCOs contained in Table 3.5-5 (e.g. Reactor Coolant System Subcooling Margin Monitor, Core Exit Thermocouples, etc.).

Section III - No Significant Hazards Evaluation

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

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

Response:

Neither the probability nor the consequences of an accident previously evaluated in the FSAR are increased since the RVLIS is a new post-TMI modification aimed at enhancing the plant's overall safety. This goal is accomplished by providing the operators with an additional advanced warning of a potential ICC condition following an accident. The proposed changes add operational criteria for RVLIS in the Technical Specifications. RVLIS does not affect the analysis of any previously evaluated accidents and decreases the consequences of small break loss of coolant accidents.

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

Response:

The possibility of a new kind of accident is not created as evidenced by the NRC Safety Evaluation Report transmitted to the Authority on May 29, 1984 which found acceptable the proposed use of the Westinghouse RVLIS. This is based on the fact that the method and manner of plant operation is unchanged. The installation of RVLIS is not an initiating event of any accident. The system is being implemented in response to NUREG-0737, Item II.F.2, and NRC Generic Letter 83-37.

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

Response:

The proposed change incorporates RVLIS into the IP-3 Technical Specifications. RVLIS will provide the operators with an additional way of detecting a potential ICC condition. Therefore, the operators' handling of an ICC condition will be enhanced and there will be no significant reduction in a margin of safety.

In the April 6, 1983 Federal Register, Vol. 48, No. 67, Page 14870, the NRC published a list of examples of amendments that are not likely to involve a significant hazards concern. Example (ii) of that list applies to the addition of RVLIS to the Technical Specifications and states:

> "A change that constitutes an additional limitation, restriction, or control not presently included in the technical specifications: for example, a more stringent surveillance requirement."

Section IV - Impact of Change

This change 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 this modification: 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) NUREG-0737, Item II.F.2
d) Generic Letter No. 83-37