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December 30, 1980
IPN-80-117

Director of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. Darrell G. Eisenhut, Director
Division of Licensing

Subject: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
Post TMI Requirements

Dear Sir:

This letter responds to your October 31, 1980 request for confirmation that the implementation dates indicated in Enclosure 1 of NUREG-0737 will be met.

Attachment 1 addresses each item of NUREG-0737 which is applicable to Indian Point Unit 3 and which the Commission has not previously identified as complete. Please note that this letter is based on the "unbound" version of NUREG-0737 which apparently was slightly revised prior to issuance of the "bound" version.

Unless otherwise indicated herein, the Authority will not propose Technical Specifications regarding these requirements until the Commission provides further clarification.

In addition, it must be noted that various items have not been sufficiently evaluated nor has engineering and design progressed to a detail whereby additional deviations can be identified at this time. Further, meeting the schedule dates indicated herein is subject to equipment availability which in most cases cannot be determined until well into the implementation period.

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The Authority will notify the Commission at the earliest possible time of any changes to the commitments provided herein.

As discussed with Mr. W. J. Ross of your staff, this letter was delayed due to the following:

•A significant amount of material was revised since the Commission's September 5, 1980 letter and each of 46 items had to be carefully reviewed.

•Many commitments required discussions with NSSS and AE contractors and involved a significant financial commitment.

•Several other licensing submittals were required by January 1, 1981 and had to be prepared concurrently.

Very truly yours,

J. P. Bayne
Senior Vice President
Nuclear Generation

Subscribed and sworn to before me this 30 day of December 1980.

Ruth G. Zapf
Notary Public

RUTH G. ZAPF
Notary Public, State of New York
No. 30-4663428
Qualified in Nassau County
Commission Expires March 30, 1982

cc: Mr. T. Rebelowski
Resident Inspector
U. S. Nuclear Regulatory Commission
P. O. Box 38
Buchanan, New York 10511

ATTACHMENT I

TMI ACTION PLAN REQUIREMENTS

SCHEDULE OF IMPLEMENTATION

OF

OUTSTANDING ITEMS

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
DECEMBER 30, 1980

I.A.1.1 - Shift Technical Advisor

1. On Duty - complete

2. Tech. Specs. - The Authority will propose Technical Specifications regarding the STA by January 1, 1981 in response to the Commission's July 2, 1980 letter regarding Technical Specifications for LL Cat. A items.

3. Trained per LL Cat. B - The Authority has completed training of all current STAs in accordance with the recommendations set forth in the INPO document, dated April 30, 1980. Final certification of these STAs is in process and will be complete by January 1, 1981. A description of this training program will be submitted to the commission by January 9, 1981.

4. Describe long term program - A longterm STA program will not differ substantially from the present program in the areas of selection criteria, qualification and training. The ultimate phase out of the STA program, however, will depend on the up-grading of shift supervisors, senior reactor operators, and reactor operators.

   If the shift supervisors become degreed engineers, the Authority will then review the position of the STA.

   This description satisfies the January 1, 1981 documentation requirement for this item.

I.A.1.3. - Shift Manning

1. Limit overtime - As stated in our October 29, 1980 letter (IPN-80-98) the Authority has revised its Administrative Procedures (AP) to meet the overtime restrictions stated in the Commission's July 31, 1980 letter. In addition, the Authority will revise the Administrative Procedures to comply with the revised overtime restrictions of NUREG-0737 to the maximum extent possible.

   The procedure will be written to minimize overtime to try to stay within the guidelines while recognizing there may be conditions where these guidelines are inappropriate.

   The procedure will contain controls such that the hours worked by key individuals will be reviewed periodically (approximately every month) to assure that excessive hours have not been assigned, in the judgment of the Resident Manager.

I.A.1.3 - cont'd

2. Minimum shift crew As noted in our October 29, 1980 response (IPN-80-98) to the Commission's July 31, 1980 letter, the Authority has revised its Administrative Procedure (AP) to meet the new shift staffing criteria. However, with regard to the additional Reactor Operator (RO) required by item (e) of the July 31, 1980 letter, the AP will reflect the following:

The additional RO is presently being provided by Consolidated Edison on an as needed reciprocal basis during Emergency Plan Conditions. This is in accordance with our response to the 90 day actions of the February 11, 1980 Confirmatory Order. This action has been reviewed by NRC staff personnel and found to be acceptable on an interim basis, provided the "on loan RO" is not allowed to manipulate any controls unless licensed on the unit.

The Authority is presently planning a Reactor Operator Training program based on criteria established in regulatory requirements as modified by Commission Orders and letters. This will allow the Authority to be in compliance with regard to the additional RO by July 1, 1982 contingent upon sufficient personnel being licensed by that date. At that time, the Authority will also submit proposed technical specifications regarding shift staffing.

I.A.2.1 - Immediate Upgrading of RO and SRO Training and SRO Training and Qualifications

The Authority has implemented each of these requirements and incorporated them into Indoctrination and Training Procedure No. 15-4, "Licensed Operator/Senior Operator Replacement Training." On October 24, 1980, the Authority submitted documentation of its compliance with item 4 (modify training).

I.A.2.3. - Administration of Training Programs

All members of the Authority training staff, including members of other organizations, who presently teach systems, integrated responses, transient, and simulator courses are SRO licensed personnel. Additional personnel who will instruct the above courses shall be licensed as SROs, or certified to teach these subjects as demonstrated by the successful completion of an SRO examination.

I.A.2.3. - Cont'd

It has been the Authority's policy to utilize senior licensed personnel as instructors prior to the Commission's March 28, 1980 letter. For this reason, these instructors participated in fullscope requalification programs as opposed to "appropriate" instructor requalification programs. It is the Authority's intent to maintain this high standard, even if future instructors are only certified, in the event that new programs are developed by the USNRC or INPO. Based on the above, a diluted requalification program has not been developed or submitted to OLB for review.

I.A.3.1. - Revise Scope and Criteria for Licensing Exams

1. and 2. Increase scope and increase passing grade - The Authority has implemented these requirements and incorporated them into Indocrination and Training Procedure No. 15-5, "Licensed Operator Requalification."

3. Simulator exams - The Authority's training program is in compliance with these requirements.

I.C.1. Short-Term Accident and Procedures Review

The Westinghouse Owners Group will submit by January 1, 1981, a detailed description of our program to comply with the requirements of Item I.C.1. The program will identify Owners Group previous submittals to the NRC, which we believe will comprise the bulk of the response. The additional effort required to obtain full compliance with this item (with proposed schedules for completion) will also be identified, as discussed with the NRC on November 12, 1980.

I.C.5 - Feedback of Operating Experience

The Authority shall implement procedures governing the feedback of operating experience to plant staff by January 1, 1981. In accordance with NUREG 0737, no documentation shall be submitted. This response supercedes the Authority's previous response to item I.C. 5 included in our June 16, 1980 letter (IPN-80-58).

I.C.6 Verify Correct Performance of Operating Activities

The Power Authority has recently received copies of the second proposed Revision 3 to Regulatory Guide 1.33 and Draft 6, October 1980 revision to ANSI/ANS 3.2-1980. Upon issuance of the final revision of ANS 3.2 we will review it to determine its impact on Indian Point 3 procedures. The following is our response to the supplemental provisions of position I.C.6 of NUREG-0737:

(1) Our procedures already incorporate the guidance of Section 5.2.6 into the surveillance test program.

(2) The authority to release systems and equipment, specifically nuclear safety-related systems, for maintenance, or return-to-service can only be granted to the senior operator on duty. This authority cannot be delegated to other personnel and assures that the shift supervisor is cognizant of plant equipment status.

With respect to surveillance testing, the on watch RO or SRO may authorize any safety related equipment surveillance test to be performed. All operating surveillance tests include steps to return the equipment to a normal mode of operation.

Any manipulation of safety-related equipment for testing maintenance and surveillance checks is done under the directions of a licensed reactor operator.

This response is in accordance with the commitments made in response to IE Bulletin 79-06A.

(3) The present Power Authority work clearance system is adequate. Before any work is performed, a work permit is filed. For similar safety related systems, work permits may only be issued by the shift supervisor on duty. All tagging of equipment is done under the direct supervision of the Shift Supervisor. Normally, an operating order is issued which controls the change of equipment status.

(4) Changes in equipment status are normally performed by the control room operators, field operators, or nuclear plant operators under the direction of licensed operators. Under certain conditions, the shift supervisor will directly initiate changes in equipment status and will inform the control room operators of such changes. Shift and relief turnover sheets ensure a continuity of knowledge of equipment status during watch rotation.

5) Normally, nuclear safety-related systems are returned to service following close-out of the Operating Order and

I.C.6 - Cont'd

either a functional retest or surveillance test is performed depending on the requirement imposed. In addition, certain safety-related systems or components are checked on either a daily, monthly or weekly basis. Spot checks are also performed by management personnel to ensure proper equipment status.

I.D.1. - Control Room Design Reviews

The Authority is currently reviewing and modifying the control room design as documented in our December 12, 1980 response (IPN-80-114) to the Commission's October 31, 1980 letter. The control room design review was initiated at Indian Point as a result of the Commission's February 11, 1980 Confirmatory Order.

I.D.2 - Plant Safety Parameter Display Console

The Authority will provide the Commission with a design description of this item ninety (90) days following issuance of final guidance by the Commission.

II.B.1 - Reactor Coolant System Vents

The Authority has purchased a Westinghouse supplied reactor vessel head vent system. The system provides the capability to vent the reactor vessel using safety related equipment upstream of the isolation valves (on the reactor side of the isolation valves).

By July 1, 1981 the Authority will provide the documentation specified in NUREG-0737 and is scheduling the installation of this system during the refueling outage in the Fall 1981. This commitment supercedes all other commitments that may have been made by the Authority on the installation of the head vent system.

II.B.2. - Plant Shielding

The Authority in its February 3, 1980 letter (IPN-80-15) reported the results of a shielding study which was undertaken to identify vital areas and equipment required for post-accident

recovery operations to ensure that accessibility was provided in accordance with the requirements of NUREG-0578 Section 2.1.6.b as clarified by the Commission's October 30, 1979 letter. Since the February 3, 1980 submittal, the Authority has been pursuing engineering associated with modifications required to provide accessibility to vital areas such as the containment isolation valves and the hydrogen recombiner panel area. One of the results of our engineering evaluation was that numerous manual valves which were required to be operated post-accident were inaccessible due to high-radiation fields and had to be automated.

Five utilities were contacted to investigate the possibility of obtaining automated valves from their current nuclear construction projects to satisfy the accessibility requirements of NUREG-0578. Some of these utilities did have a few available automated valves, however, none of the valves could be adapted to satisfy IP-3 requirements.

A parallel valve search effort was directed at nuclear-qualified valve manufacturers. Twelve companies were contacted. Of these, none were able to deliver the required motorized valves by the October 1980 outage.

An order was, therefore, placed for thirty-two (32) motor operated valves with a tentative delivery date of July 6, 1981.

The Authority will make every effort to complete installation by the specified date of January 1, 1982 but feels that realistically all the modifications associated with Item No. II.B.2 cannot be completed until our Spring 1983 refueling outage.

Based on review of the clarification of this item in NUREG-0737, the radiation sources have been significantly reduced. Upon completion of our evaluation of the modifications discussed above in light of the NUREG-0737 clarification, the Authority will notify the Commission of any changes to the above schedule.

The Authority has submitted documentation regarding environmental qualification of safety-related electrical equipment in accordance with the Commission's NRC Bullentins 79-01, 01A, 01B, Supplements 1,2,& 3 and the September 19, 1980 Order. Based on the Clarification of NUREG-0737, the Authority considers environmental qualification to be limited to safety related electrical equipment.

This response supercedes our previous response to lessons learned short-term item 2.1.6.b in our February 3, 1980 letter, (IPN-80-15).

II.B.3 - Postaccident Sampling

The Authority will comply with the NUREG-0737 clarification under the following conditions:

1) Clarification item 2b - the containment atmosphere hydrogen monitor shall be provided in accordance with item II.F.1.(6) of NUREG-0737.

2) Clarification item 9a - the Authority presently has the capability to analyze sample concentrations of up to 10 ci/g. However, the system is designed to provide the shielding required to meet 10 CFR 50, General Design Criteria (GDC) personnel exposure requirements based on the originally specified concentration of .5 ci/g. The Authority does not believe that the Commission requires the sampling system to be redesigned to accommodate samples with concentrations up to 10 ci/g while still meeting the GDC limits for personnel exposure.

II.B.4 - Training for Mitigating Core Damage

As a result of the efforts of a number of sources, among them the Institute of Nuclear Power Operations (INPO) and the Westinghouse Owners Group, procedures are continually being developed and issued which deal with the subject of mitigating core damage. As these procedures are received and reviewed, changes are made to IP3 plant procedures which reflect the latest information on this subject. To date, all Authority licensed personnel are trained in the use of these procedures, as they are part of our normal requalification training program. As new and pertinent information becomes available, the Authority will continue to update its procedures and the subject matter for requalification training in this regard.

This program of training will be available for review by January 1, 1981, at whatever stage of development such industry-wide information is at that time. Completion of this training is expected by October 1, 1981, in accordance with the guidelines of Enclosure 3 to the Commission's March 28, 1980 letter.

II.D.1. - Relief and Safety Valve Test Requirements

As a sponsor of the EPRI PWR Safety and Relief Valve Test Program, the Authority intends to comply with the requirements

of NUREG-0578, Item 2.1.2. By letter dated December 15, 1980, R.C. Youngdahl of Consumers Power Company has provided the current PWR Utilities' positions on NUREG 0737, Item II.D.1 clarifications. Briefly those positions are:

A. Safety and Relief Valves and Piping the EPRI "Program Plan for Performance Testing of PWR Safety and Relief Valves", Revision 1, dated July 1, 1980, does provide a program that satisfies the NRC requirements. Discussions with the NRC staff and their consultants are resolving specific detailed issues.

B. Block Valves - The EPRI Program has not formally included the testing of block valves. However, a small number of block valves have been tested at the Marshall Steam Station Test Facility. The PWR Utilities and EPRI can not provide a detailed block valve test program until results of the Wyle and CE relief valve tests are available. Therefore, a block valve test program will not be provided before July, 1981. The PWR Utilities and EPRI believe that the proper operation of the TMI-2, and Crystal River block valves and other operational experience, plus knowledge of the Marshall tests, support a less hurried and more rational approach to block valve testing.

C. ATWS Testing - PWR Utilities will not support additional efforts for ATWS valve testing until regulatory issues are resolved. The major safety and relief valve test facility (CE) is nearing completion and some measures were taken to provide additional test capability beyond the current program requirements. The NRC should recognize that results from the current program are likely to provide most of the information necessary to address ATWS events (i.e., relief capability at high pressures).

II.D.3. - Valve Position Indication

Implementation of this item is complete. The environmental and seismic qualification program began on August 15, 1980 and presently is in its final phases. Upon completion of the program, tentatively scheduled for January 1, 1981, a qualification document will be prepared to verify that the tested equipment is qualified for its intended service in accordance with IEEE-323-1974 and IEEE-344-1975.

The Authority will propose Technical Specifications regarding this item by January 1, 1981 in response to the Commission's July 2, 1980 letter.

II.E.1.1 - Auxiliary Feedwater System (AFWS) Evaluation

The Authority has completed its evaluation of the AFWS as documented in letters dated August 11, 1980 (IPN-89-77) and November 26, 1980 (IPN-80-112) which were submitted in response to the Commission's February 11, 1980 Confirmatory Order and October 9, 1980 letter, respectively.

II.E.1.2 - Auxiliary Feedwater System

1. Automatic Initiation - The following response addresses each of the paragraphs of IEEE 279-1971 referenced in NUREG-0737:

   4.1 General Functional Requirements - Automatic initiation of the protective action is required when the condition monitored reaches a preset point. The protective function of this system is to maintain water inventory in the Steam Generators. To accomplish this function the system is designed to have the two motor pumps or the turbine driven pump feed to the Steam Generators. The two motor driven pumps receive automatic starts derived from the Engineered Safeguard Circuits. All three pumps receive start signals from the Low-Low Steam Generator Level Circuity. The steam driven pump must be throttled manually in order to bring the pump up to speed and prevent pump damage. The Authority believes that the intent of section 4.1 is met.

   4.2 Single Failure - The Authority has conducted a review of the Auxiliary Feedwater System and found application of single failure analysis yields no violation of that criteria.

   4.3 Quality of Components and Modules - The auxiliary feedwater system (AFWS) was originally specified as safety related. At the time of construction, the Quality Control Program had applied to components of the AFWS. Consolidated Edison owned and operated IP-3 prior to the Authority and considered the AFWS a safety-related system thus falling within their Quality Assurance Program. This program ensured the highest quality and control of the system. The Power Authority, after purchasing the plant from Consolidated Edison, also considered the system safety related and followed in the tradition of closely controlling the quality of the Auxiliary Feedwater System.

   4.4 Equipment Qualification - The following is a list of equipment that is associated with the Auxiliary Feedwater System. Most of this information was submitted to the NRC on April 28, 1980 and November 1, 1980 in response to I.E. Bulletin 79-01B and the Commission's September 19, 1980 Order:

TRANSMITTERS				Qual. Method	REF. & Notes
PT	1126	Foxboro E11GM	Steam Press. to Aux. FP	Analysis	d,r
PT	1205	Foxboro E11GM	City H_2O Supply to Aux. FP	Analysis	e,r
FT	1200				
	1201				
	1202				
	1203	Foxboro E13DM	Aux. F.W. Flow	Analysis	e,r
PT	1163				
	1164				
	1165				
	1166	Foxboro E11GM	S.G. Feedwater Press.	Separate Analysis	e
PT	406A				
	406B	Foxboro E11GM	AFP Discharge Pressure	Analysis	e,r
PT	1263				
	1264				
	1265	Foxboro E11GM	AFW Pump Suction	Analysis	e,r
PT	1260				
	1261				
	1262	Foxboro E11GM	AFW Pump Discharge Press.	Analysis	e,r

			Qual. Method	REF. & Notes
SOLENOID VALVES				
FCV1121 1123	ASCO 6300	AFW Recir. Flow	Analysis	e,f,r
LCV1158	ASCO 8317	Cond. Storage Tank Valve	Analysis	4,5,f
PCV1187 1188 1189	ASCO 8300	City Water Suction Control Valve	Analysis	b,r
PCV1139	ASCO 8300	AFW Pump Steam Press. Valve	Analysis	d,r
PCV1310 A&B	ASCO 8316	Steam Supply A.F.P. Turbine	Analysis	6
FCV1121 1123	NAMCO EA170	AFW Recir. Flow	Analysis	e,r,u

PCV1310 A&B	MICRO-Switch 7313 Micro-Switch	Steam Supply to AFW Turbine AFW Pump Steam	Analysis	u
PCV1139	OPAR	Press. Valve	Analysis	e,r,u
PCV1187 1188 1189	Micro-Switch BZE-2RN	City Water Suction Control Valve	Analysis	e,r,u

MOTORS

Auxiliary Feedwater Pump Motor	Frame No. 509 US	Analysis	e,r
Electro-Automatic Transducers	Auxiliary Feedwater	Analysis	e,r

FCV405,A,B,C,& D Fisher-Governor Flow Control Valve

FCV406,A,B,C,& D

CABLE

Shielded Twisted Pair and QUAD Instrument Cable	Lewis Engineering Co.	(113-9)	e,r
Single Conductor Power Cable	General Electric Co.	(113-8)	e,r
Multi-Conductor, Control & Small Power Cable	General Electric Co.	(113-7)	e,r

TEMPERATURE SENSORS

Auxiliary Feedwater Pump #32	Analysis	AA

NOTES

(b) Pressure and temperature changes approximate ambient conditions.

(d) Function not required if auxiliary feedwater pump steamline breaks.

(e) Conditions are near normal during the accident. Area temperature and pressure return to normal within 5 minutes due to sensors set at 135°F causing auxiliary feedwater pump steamline isolation.

(f) Valve will close (safe position) for all potential modes of failure.

(r) A concurrent break in the main steam line, feedwater system or a loss of coolant accident in conjunction with a break in the auxiliary feedwater pump steam line results in a double failure and is not considered using single failure analysis. A break in the auxiliary feedwater pump steam line downstream from valves 1310 A & B, which are located outside the auxiliary feedwater pump room, is the accident considered. With this break, the Auxiliary Feed Water System (AFWS) may not be initiated because of the small loss of steam and the subsequent isolation of this line by the closing of valves 1310 A & B. This will maintain the design environmental conditions for the equipment that is associated with the AFWS. Therefore, this system is still functional based on the consequences of this accident.

NOTES - Cont'd

(u) Limit Switch is for position indication only.

(AA) The Authority is currently verifying the qualification and will report the results under our environmental qualification responses.

REFERENCES

4. Westinghouse Electric Corporation, letter (WS-CE-755). Potential mode of failure identified for solenoid valves, C. Ficheldinger to Vinsalo, August 15, 1975.

5. ASCO solenoid valves contain nylon with a sensitivity of 5 x 107. Reactor Shielding Design Manual, first edition, D. Van Norstand Company Inc., ("Group 7" Plastics). Buna N with a sensitivity of a 6 x 10^6 (per ASCO Company).

4.6 Channel Independence - The original D.C. Power System is such that 3 station batteries are called upon to feed 4 channels of components. Channel III and IV are assigned to the same battery and some overlap between these channels is permissible. The two motor driven pumps which make up one actuation train occupies channel I, III and IV. The other train associated with the turbine driven pump occupies channel II. Generally the system has no channel independence concerns with the exception of steam supply valve 1310B to the turbine driven pump. This valve is powered from channel 1 which is the same channel associated with motor driven pump No. 33. However, since the 1310B valve fails open and does not involve a large amount of instruments and control associated with it the Authority believes the intent of section 4.6 is met.

4.7 Control and Protection Interaction - The control and protection circuity involves Steam Generator level, safety injection initiation, Main Boiler Feed Pump controls, blackout initiation and breaker controls. Various isolation devices exist to prevent interactions such as contact to coil and contact to control isloations or no boundary exists, such as, for the breaker controls.

4.9 Capability For Sensor Checks - The auxiliary feedwater pumps are started via the following signals:

1) Safety Injection Train A

2) Safety Injection Train B

3) Low-Low Steam Generator Level

4) Black Out (No Safety Injection)

5) The Main Boiler Feed Pumps Trip

Signal (5) is a non-safety related trip function, and, therefore, is not subjected to the requirements of sensors checks. The engineered safeguards Train A and B are a result of various matrices which are monitored routinely and therefore, do not lend themselves to the requirements of sensor checks. The steam generators levels are monitored routinely and compared to determine anomolies, and therefore meet the requirements of sensor checks. Finally, the non-engineered safeguards blackout signal is actuated by a high-speed relay which does not lend itself to a meaningful comparison. Although the control valves do provide the path to the steam generators from the auxiliary feed pumps, it is felt that the paragraph is not applicable.

4.10 Capability for Test Calibration - As stated in our response to NUREG-0578, testing and capability exists for the steam generator level and black-out feature of automatic initiation. the main boiler feed pump trip is non-safety related and therefore, does not fall within the requirements of section 4.10.

4.11 Channel Bypass or Removal From Operations - The Auxiliary Feedwater System channels associated with low-low steam generator level can be removed from service for maintenance, testing or calibrations. This feature will not initiate a protective function and yet meets single failure criteria and the minimum number of channels required by our technical specifications. There are no bypasses of a channel nature, therefore the intent of this paragraph is met.

4.12 Operation Bypasses - No operating bypasses exist on a system protective function level, therefore, the requirements of this paragraph are met.

4.13 Indication of Bypasses - No bypass exist therefore, the section is not applicable.

4.17 Manual Initiation - this paragraph requires provisions for manual initiation of the protective action on the system level. Provision exists for manual initiation on the component level, but no provision exists for manually initiating the Auxiliary Feedwater System as a whole.

It is worth noting that since the motor driven pumps flow paths are aligned for automatic actuation and the breaker controls are in proximitry, manual actuation pushbutton would not be beneficial.

The Authority believes the design of the Auxiliary Feedwater System is adequate, and do not plan any plant modifications.

2. Flowrate Indication

As stated in our response to NUREG-o578, we presently have a flowmeter and wide-range steam generator level indication for each steam generator.

A. Specific Requirements:

I) Environmentally qualified: The environmental qualification of the auxiliary feedwater flow indication is discussed in Part 1 of this response. The steam generator level transmitter will be replaced with the IEEE qualified transmitters at an outage of sufficient duration after they are received. The environmental qualification of the existing transmitters are discussed in our responses to Bulletin 79-01 and related responses.

II) As stated in our response to NUREG-0578 the flow indication system is powered from a vital instrument bus which is battery-backed. The Authority does not understand why a "Non-Class IE power source" is required. Our power system is considered safety-related.

III) As stated in our response to NUREG-o578, testability exists for the auxiliary feedwater flow and steam generator level instrumentation.

IV) The auxiliary feedwater flow and steam generator level systems fall under our quality assurance program.

V) The displays for auxiliary feedwater flow and steam generator level is a continuous reading indicator.

B. The use of the auxiliary feedwater flow indicators and steam generators are incorporated into emergency procedures and operator training programs under normal and emergency conditions. Currently, the Authority employs a consultant to review our control room indication and alarm system from a human engineering point view. The Authority is in the process of reviewing their recommendations, and will consider implementation of same during a refueling outage.

The Authority will propose Technical Specifications regarding the Auxiliary Feedwater System Initiation and Flowrate Indication by January 1, 1981 in response to the Commission's July 2, 1980 letter.

The above responses constitute the documentation required by January 1, 1981 for item II.E.1.2.

II.E.3.1 Emergency Power for Pressurizer Heaters

Implementation of this item is complete and the Authority will propose Technical Specifications regarding this item by January 1, 1981 in response to the Commission's July 2, 1980 letter.

II.E.4.2. - Containment Isolation Dependability

Positions 1-4

The Authority complies with NRC positions 1-4 except as follows:

Position 3 states that all nonessential systems shall be automatically isolated by the containment isolation signal. Five utilities were contacted to investigate the possibility of obtaining automated valves from their current nuclear construction projects to satisfy the isolation requirements of NUREG-0578, Section 2.1.4. Some of these utilities did have a few available automated valves, however, none of the valves could be adapted for IP3 application.

A parallel valve search effort was directed at nuclear qualified, valve manufacturers. Twelve companies were contacted of which two (2) companies were able to deliver four (4) out of the seven (7) required air-operated valves within an acceptable period of time to support installation during the planned outage of October, 1980. The following nonessential, air operated, containment isolation valves have, therefore been installed with the required Phase A isolation circuitry:

N_2 valves No. 1610 and 550, previously manual. Demineralized System Valves Nos. DW-1 and DW-2 (new valves).

The three remaining air-operated valves could not be delivered on time for the October 1980 installation but have been purchased with a July 6, 1981 tentative delivery date. These are nonessential, air-operated valves for the Sampling System to replace manual valves 958, 990A and 990B. The valves have been purchased on IP-PO-549 and will be installed during the scheduled Fall 1981 refueling outage.

As stated in our February 3, 1980 letter (IPN-80-15) all other manual non-essential valves are locked shut during normal operations. In the interim, for valves which still need to be automated, the plant operating procedures re-emphasize to the operating personnel that these valves shall not be opened unless absolutely essential for plant operations or testing, that an operator will be dedicated to the operation of these valves as long as they are in the open position and that their first response to any emergency condition while any of these are open shall be to insure that the valves are immediately returned to the closed position.

Position 5 - Containment Pressure Setpoint

The present containment pressure setpoint for initiation of engineered safeguards is 3 psig. Pressure relief of containment is initiated at approximately 1 psig. The system accuracy is approxmiately ±2 psig, therefore, the existing setpoint cannot be lowered any further. It is the Authority's position that lowering the setpoint any further will cause spurious challenges to the engineered safeguards systems during plant operations. This position is consistant with that of the Westinghouse Corporation.

Position 6 - Containment Purge Valves

The purge valves are maintained in the closed position when the plant is above the cold shutdown condition. An operating procedure has been written to reflect this requirement until such time that all the concerns of BTP-6-4 and the Staff's interim position of October 23, 1979 have been satisfactorily resolved. By January 1, 1981, the purge valves will be mechanically or electrically prevented from being inadvertantly opened. An adequate method will be used to verify every 24 hours that the purge valves will be closed when required to be closed.

Position 7 - Radiation Signal on Purge Valves

As required by this position the purge and vent valves close on a high radiation signal.

The Authority will address Technical Specifications regarding Containment Isolation by January 1, 1981 in response to the Commission's July 2, 1980 letter.

II.F.1.(1) Noble Gas Monitor

The Authority will comply with the NUREG-0737 clarification for the noble gas monitor. The monitor design details will be available by January 1, 1981. Installation design details will be available at a later date.

II.F.1.(2) - Iodine/Particulate Sampling

The Authority will comply with the NUREG-0737 clarification under the following conditions:

1) Clarification item 3 - Due to the prefiltration of particulates in the vent stack, IP3 has a known small particulate size (i.e. .3 microns). Therefore, substantial deviations from isokinetic velocities can be tolerated without losing sample validity. In all cases sampling systems will comply with ANSI N13.1 (1969).

II.F.I.(2) - Cont'd

2) Clarification item 4 - The Authority will not sample the condenser air ejector discharge since the plant instrumentation provides for discharging the air ejector effluent to the containment upon a high radiation signal by the air ejector monitor.

Final design details of the implementation of this item will be available for review by January 1, 1981.

II.F.1 (3) - Containment High Range Monitor

The Authority will comply with the NUREG-0737 clarification. Final design details will be available for review by July 1, 1981.

II.F.1.(4) - Containment Pressure Monitor

The Authority is in the process of installing two safety related pressure transmitters in the pipe penetration area and two safety related recorders with accessory electronic devices in the control room.

Conformance to Action Plan "Clarification"

1. The pressure channels meet the design and qualification criteria prescribed in Table 1 of Draft Reg. Guide 1.97 Rev. 2 dated June 4, 1980 for category 1 instruments.

2. Measurement and indication capability is from 10 PSIA to 200 PSIG. IP3 is not a subatmospheric containment and therefore measurement to 5 PSIA is not applicable. The design pressure for the IP3 concrete containment is 47 PSIG; three times the design pressure is 141 PSIG, well within the 200 PSIG range of the pressure channels.

3. The overall accuracy of the pressure measurement channel is less than ±1%, well within the mandatory accuracy of ±10%.

4. A continuous display of two pressure measurement channels is recorded in the control room.

5. The total response time of the pressure transmitter is less than 200 miliseconds. However, the response time of the state-of-the-art recorder is 3 seconds for travel from 10% to 90% of scale.

Status All equipment has been delivered to the site and installation is approximately 50% complete. The Authority feels that the schedule specified in NUREG-0737 of January 1, 1982 for implementation will be met. This commitment supersedes any other commitments that may have been made by the Authority previously regarding this item.

II.F.1.(5) - Containment Water Level Monitor

The Authority is in the process of installing six safety related water level measurement channels including two channels measuring RHR Recirculation Sump level, two channels measuring Containment Sump level and two channels measuring Containment Building level. Level transmitters will be located inside the containment building; safety related recorders and electronic devices will be located in the control room. Electronic signals are transmitted from the containment building to the control room through existing electrical penetrations.

Conformance to Action Plan "Clarification"

1. All six (6) level channels meet the design and qualification criteria prescribed in Table 1 of Draft Reg. Guide 1.97, Rev. 2, dated June 4, 1980 for category 1 instruments.

2. The two (2) Containment Building level transmitters will be capable of measuring water level to Elevation 52'-4". The available water which could flood the containment is the 360,000 gallons from the Refueling Water Storage Tank; this quantity of water would result in flooding to Elevation 49'-4". A quantity of 500,000 gallons would result in flooding to Elevation 51'3" ±; 600,000 gallons would result in flooding to Elevation 53'-0"± . All level transmitters are located at elevation 54'-0" or above.

3. During normal operation the overall accuracy of each level measurement channel is less than ± 1%, well within the ± 5% limit set by the Commission. However, based on preliminary information, the accuracy could deteriorate due to LOCA conditions; the temperature effect on the transmitter could be ±10%; the radiation effect on the transmitter could be ±5%. The Authority is proceeding with implementation of this instrumentation since it is the best available at this time. Final test reports and certificates of compliance will be available during January, 1981.

It should also be noted that six (6) level measurements will produce a recorded readout in the control room when the containment building water level is in the critical range of 4" to 28" above the containment floor of Elevation 46'0"; four recorded readouts are available from the bottom of the sumps to 4" above the containment floor. This large number of measurements and readings will provide the operator with an extra confidence level to ascertain a valid reading during accident conditions even though the transmitter accuracy may be affected to a greater degree than desired.

Status - All equipment except Barton Transmitters have been delivered to the site. Most of the cable and conduit installation is complete. Piping and supports are approximately 50% complete. The Authority feels that the schedule of January 1, 1982 specified in NUREG-0737 for implementation will be met.

II.F.1.(6) - Containment Hydrogen Monitor

The Authority will install two safety related Hydrogen Concentration Measurement Cabinets in the Primary Auxiliary Building, Elevation 41'0", and two safety related recorders with accessory electronic devices in the control room.

The influent lines to the Hydrogen Concentration Measurement Cabinets will have as their source point the five existing sampling lines passing through the containment penetrations to each fan unit plenum. The five sampling lines along with two return lines are part of the present Postaccident Containment Sampling System which will be retired.

Two of the five sampling lines will be routed to a common source line and then to a hydrogen monitor; the three remaining sampling lines will be likewise headered and routed to the remaining hydrogen monitor. Each monitor has a separate return line.

Redundant containment isolation valves shut-off both the five sampling lines and the two return lines; two pressurization air shut-off valves supply air to each space between two containment isolation valves for sealing purposes.

Conformance To Action Plan "Clarification"

1. The Hydrogen Concentration Measurement channels meet the design and qualification criteria prescribed in Table 1 of Draft Reg. Guide 1.97, Rev. 2, dated June 4, 1980 for Category 1 instruments.

2. During normal operation the Hydrogen Concentration Measurement Cabinets are kept in the "STANDBY" mode; the containment isolation valves are closed. To satisfy the requirement of a hydrogen concentration reading 30 minutes after safety injection, the operating personnel in the control room must activate the system by opening the appropriate isolation valves and starting up the sample pump. All of these controls are located at the Control Room Cabinets. The Authority will verify that a containment building air sample requires less than 30 minutes to travel from the containment building to the analyzer cabinets.

3. The overall measurement accuracy is within ± 2.50% of range for the entire measurement channel. We are proceeding with implementation of this instrumentation since it is the best available at this time.

STATUS - All equipment except the hydrogen analyzers (Comsip-Delphi model K-III) have been delivered to the site. Most cable and conduit has been installed. Piping installation is approximately 25% complete. The Authority feels that the schedule for installation specified in NUREG-0737 will be met. This commitment supercedes all other commitments that may have been made by the Authority previously regarding this item.

II.F.2. - Instrumentation for Detection of Inadequate Core Cooling

1. Subcooling meter - The Authority has completed installation of the analog system discussed in our February 3, 1980 letter (IPN-80-15). This system will automatically choose the conservative reactor coolant system (RCS) pressure out of two inputs and the conservative temperature out of four inputs and then compute the saturation margin using these parameters. The Authority will not utilize the analog system until Commission approval is received.

2. Tech. Specs. The Authority will propose Technical Specifications for the subcooling meter by January 1, 1981 in response to the Commission's July 2, 1980 letter.

3. Install Level Instrumentation -The Authority is actively pursuing the design and installation of a Reactor Vessel Level Measurement System to provide a means to monitor the water level in the reactor vessel for detection of inadequate core cooling.

   The Authority is concerned with the capability of the Westinghouse design, recently submitted to the Commission. Other designs are available which may be superior to the

II.F.2. - Cont'd

Westinghouse design. Upon completion of our evaluation of these designs the Authority will choose a vendor. Following this decision the Authority will submit a design report in accordane with NUREG-0737.

At this time, it appears unlikely that installation of this item can be accomplished by January 1, 1982. The Authority will provide the Commission with a schedule for installation along with our design report.

II.G.1. - Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators

Implementation of this item is complete and the Authority will propose Technical Specifications regarding this item by January 1, 1981 in response to the Commission's July 2, 1980 letter.

II.K.2.13 - Orders on B&W Plants; Thermal - Mechanical Report

To completely address the NRC Requirments of detailed analysis of the thermal-mechanical conditions in the reactor vessel during recovery from small breaks with an extended loss of all feedwater, a program will be completed and documented to the NRC by January 1, 1982. This program will consist of analysis for generic W PWR Plant Groupings. Following completion of this generic program, additional plant specific analyses, if required, will be provided. A schedule for the plant specific analysis will be determined based on the results of the generic analysis.

II.K.2.17 - Order on B&W Plants; Voiding in RCS

The Westinghouse Owners Group is currently addressing the potential for void formation in the Reactor Coolant System (RCS) during natural circulation cooldown conditions, as described in Westinghouse letter NS-TMA-2298 (T.M. Anderson, W. to P.S. Check, NRC). We believe the results of this effort will fully address the NRC requirement for analysis to determine the potential for voiding in the RCS during anticipated transients. A report describing the results of this effort will be provided to the NRC before January 1, 1982.

II.K.2.19 - Orders on B&W Plants; Benchmark Analysis of Sequential AFW flow

The transient analysis code, LOFTRAN, and the present small break evaluations analysis code, WFLASH, have both undergone benchmarking against plant information or experimental test facilities. These codes have also been compared with each other under appropriate conditions. The WOG will provide a report addressing the benchmarking of these codes on a schedule consistent with the requirements of task II.K.2.19.

II.K.3.1 - Automatic PORV Isolation, and
II.K.3.2. - Report on PORV Failures

The Westinghouse Owners Group is in the process of developing a report (including historical valve failure rate data and documentation of actions taken since the TMI-2 event to decrease the probability of a stuckopen PORV) to address the NRC concerns of Item II.K.3.2. However, due to the time-consuming process of data gathering, breakdown, and evaluation, this report is scheduled for submittal to the NRC on March 1, 1981. This response supercedes our June 16, 1980 response (IPN-80-58) regarding this item.

II.K.3.3 - Reporting of SV and RV Failures and Challenges

The Authority does and will continue to promptly report failures of pressurizer safety and relief valves to the NRC in accordance with the IP3 Technical Specifications.

Since April 1, 1980, there have been no challenges to pressurizer relief or safety valves and no failures of pressurizer relief valves. There has been only one failure of a safety valve which occured on October 17, 1980 (LER-80-013/03L-0, dtd November 15, 1980; IPP-FWG-10850). The safety valve would not fully pop open within its required set pressure during a refueling surveillance test.

The above discussion constitutes the Authority's 1980 report on this subject. Therefore, the Authority will not submit another report on this subject to the Commission until the end of 1981.

This response supercedes the Authority's previous response to item II.K.3.3 in our June 16, 1980 letter (IPN-80-58).

II.K.3.5 - Auto Trip of RCPs

The Westinghouse Owners Group resolution of this issue has been to perform analyses using the Westinghouse small break evaluation model

II.K.3.5 - Cont'd

(WFLASH) to show ample time is available for the operator to trip the reactor coolant pumps following certain size small breaks (see WCAP-9584). In addition the Owners Group is supporting a best estimate study using the NOTRUMP computer code to demonstrate that tripping the reactor coolant pump at the worst trip time, after a small break will lead to acceptable results.

For both of these analysis efforts, the Westinghouse Owners Group is performing blind post-test predictions of LOFT experiment L3-6. The input data and model to be used with WFLASH on LOFT L3-6 was submitted to the Staff on 12/1/80 (NS-TMA-2348). The information to be used with NOTRUMP on LOFT L3-6 will be submitted prior to performance of the L3-6 test as stated in letter OG-45, dated 12/3/80.

The LOFT prediction from both models will be submitted to the staff on February 15, 1981 given that the test is performed on schedule. The best estimate study is scheduled for completion by April 1, 1981.

Based on these studies, the Westinghouse Owners Group believes that resolution of this issue will be achieved without any design modifications. In the event that this is not the case, a schedule will be provided for potential modifications. This response supersedes the Authority's previous response to this item in our June 16, 1980 letter (IPN-80-58).

II.K.3.9 - PID Controller

As stated in our November 16, 1980 letter (IPN-80-110), the Authority has completed the Westinghouse recommended modifications to the PID controller.

II.K.3.10 - Proposed Anticipatory Trip Modification

As stated in our June 16, 1980 letter (IPN8058), the Authority, to date, has not proposed a modification of the anticipatory trip (i.e., turbine trip above 10% power results in a reactor trip). However, should such a modification be proposed in the future, it will be documented and submitted to NRC for approval.

II.K.3.12 - Confirm Existence of Anticipatory Trip Upon Turbine Trip

As stated in our June 16, 1980 letter (IPN-80-58), Indian Point Unit 3 has an anticipatory trip upon turbine trip when reactor power is above 10%. Therefore, no further action is required.

II.K.3.17. - ECC System Outages

In accordance with our June 16, 1980 letter (IPN-80-58), the following report is provided which details the maintenance outages and testing of ECCS components under the following conditions:

a) The period investigated was from initial criticality, April 1976, to the present, December 1980.

b) Only outages of components which placed the plant in a limiting condition for operation allowed by the Technical Specifications were considered. Maintenance and testing of components when not required to be operable with respect to Technical Specifications have not been included.

c) The equipment investigated was that identified as ECCS equipment in the plant's FSAR and SER.

d) Only testing during plant operation which prevented the component from performing its intended function was considered.

The following components were identified as having maintenance outages:

a)	Component	- Boron Injection Tank
	Outage dates	- 9/19/76 to 9/30/76
	Duration	- 262 hours
	Cause	- Corroded weld on outlet line
	Corrective Action	- Defective weld repaired and tested
b)	Component	- No. 31 High Head Safety Injection Pump
	Outage dates	- 4/4/77 to 4/7/77
	Duration	- 68 hours
	Cause	- Low Discharge pressure
	Corrective Action	- Replaced pump internals and pump tested

As a result of increasingly stringent testing requirements, the Injection Pump test was revised in April 1980 and now requires the discharge valves of one pump to be closed during periodic testing. The following table indicates the dates and the approximate duration of testing of this pump up to 12/15/80:

II.K.3.17 - Cont'd

DATES	S.I. PUMP NO. 32 TESTING DURATION
5/16/80	10 Minutes
6/15/80	10 Minutes
7/11/80	10 Minutes
8/15/80	10 Minutes
9/12/80	10 Minutes
10/7/80	10 Minutes
11/4/80	10 Minutes
12/14/80	10 Minutes
TOTAL TIME	80 Minutes

Based on the results of this investigation, there are no proposed changes to the Technical Specifications for the ECC Systems at this time.

II.K.3.25 - Power on Pump Seals

The Indian Point 3 Facility has an automatic start feature of the component cooling water system on sequences requiring this system to be operable. One exception (safety injection coincident with a loss of offsite power) to the auto start feature is governed by operator action as referenced in plant emergency procedures.

Relative to pump seal failure, the results of any analysis performed and proposed modifications will be forwarded to the Commission as addressed in NUREG-0737. The date for submittal will be 1/1/82 with appropriate modifications, if any, implemented by 7/1/82.

II.K.3.30 - Small Break LOCA Methods

As stated in our June 16, 1980 letter (IPN-80-58), the present Westinghouse Small Break Evaluation Model used to analyze Indian Point Unit 3 is in comformance with 10 CFR Part 50, Appendix K. However, Westinghouse has indicated that they will, nevertheless, address the specific NRC items contained in NUREG-0611 in a model change scheduled for completion by July 1, 1983.

II.K.3.31. - Compliance with 10 CFR 50.46

The Authority's response to this item is the same as that previously documented in our June 16, 1980 letter (IPN-80-58) which follows:

The present Westinghouse Small Break Evaluation Model and Small Break LOCA Analyses for Indian Point Unit 3 are in conformance with 10 CFR Part 50, Appendix K and 10 CFR Part 50.46. As stated in response to Item II.K.3.30, Westinghouse plans to submit a new Small Break Evaluation Model to the NRC for review by July 1, 1983.

If the results of this new Westinghouse model (and subsequent NRC review and approval) indicate that the present small break LOCA analysis for Indian Point Unit 3 are not in conformance with 10 CFR Part 50.46, a new analysis utilizing the new and approved Westinghouse model will be submitted to the NRC in accordance with the NRC schedule.

III.A.1.2. - Upgrade Emergency Support Facilities

The Authority will provide the Commission with design details of this item ninety (90) days following issuance of final guidance by the Commission.

III.A.2 - Emergency Preparedness

1. Upgrade Emergency Plan - The Authority is confident that the NUREG-0737 schedule will be met, with the exception of the public notification system required by July 1, 1981. Vendor proposals should be solicited by January 1, 1981 and completion of the system will depend upon equipment lead time which the selected vendor will require.

   The Authority will propose a license condition regarding the Emergency Plan by January 1, 1981 in the response to Commission's July 2, 1980 letter.

2. Meteorological data - The Authority is confident that the staged implementation schedule required by NUREG-0737 will be met. This conclusion is based on information received from Consolidated Edison Company which is the lead utility for both Indian Point Units 2 and 3 with respect to the Meteorological program.

III.D.1.1 - Primary Coolant Outside Containment

Implementation of this item is complete and the Authority will propose a license condition regarding this item by January 1, 1981 in response to the Commission's July 2, 1980 letter.

III.D.3.3 - Inplant Radiation Monitoring

The Authority presently complies with the NUREG-0737 requirements regarding this item. Compliance was verified by the Commission in March 1980. Based on the NUREG-0737 clarification, no documentation is required to be submitted.

III.D.3.4. - Control Room Habitability

The Authority will submit a report by January 16, 1981, which will include the information requested in NUREG-0737, item III.D.3.4, attachment 1, with the exception of subsections (4) and (5).

Information regarding item (4) will be submitted to the Commission upon completion of a toxic gas hazards survey scheduled for the near future.

The Technical Specification requirements referred to in NUREG-0737 will not be proposed since the Authority has determined that a chlorine detection system and an independent air filtration system are not required at this time. Justification for this position will be documented in the above mentioned report.

This response supercedes our June 16, 1980 response (IPN-80-58) regarding this item.