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November 21, 1979
IPN-79-83

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

Attention: Mr. Albert Schwencer, Chief
Operating Reactors Branch No. 1
Division of Operating Reactors

Subject: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
Short Term Requirements of TMI Lessons Learned

References:
(1) Letter, Darrell G. Eisenhut (NRC) to All Operating Nuclear Power Plants, dated September 13, 1979
(2) Letter, Paul J. Early (PASNY) to Darrell G. Eisenhut (NRC), dated October 22, 1979 (IPN-79-75)
(3) Letter, Harold R. Denton (NRC) to All Operating Nuclear Power Plants, dated October 30, 1979

Gentlemen:

Enclosed as Attachment 1 to this letter are commitments and responses beyond those provided in Reference (2).

These commitments and responses are provided based upon the clarification of NRC requirements contained in Reference (3), and in a phone conversation with the NRC staff on November 13, 1979.

Very truly yours,

J. R. Schmieder
J. R. Schmieder
Executive Vice President
and Chief Engineer

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ATTACHMENT 1

THREE MILE ISLAND LESSONS LEARNED COMMITMENTS

Section 2.1.1

Pressurizer Heaters

The pressurizer heaters, one control group and three back up groups, are all powered from a separate safety related 480 volt bus. The heaters will be stripped from that bus during a safety injection (SI) or loss of bus voltage. After resetting S.I. their power supply can be re-established manually from the control room. The appropriate operating procedure shall highlight this point. The 480 volt safety buses and diesels are sized for full safeguard loads during a DBA LOCA. For smaller breaks, all the safeguards equipment would not be required and redundant components would be secured thus providing the needed capacity for the energizing of the pressurizer heaters. The pressurizer heaters have safety related circuit breakers for main and control power.

One group of heaters are required to maintain natural circulation. This point will also be highlighted in the appropriate operating procedure which will be revised by January 1, 1980.

Pressurizer Level and Relief Block Valves

The pressurizer PORV's solenoids are powered from the battery supplied 125 v DC system. The valves use nitrogen as a motive force via solenoids. The nitrogen is stored in accumulators and tanks which are independent of offsite power. The block valves are powered from a vital 480 v motor control center which obtains power from the emergency diesels in the event of loss of outside power. This changeover is accomplished automatically. The design of the PORV's and block valves are such that they can be opened in addition to being closed in the event of a loss of offsite power. The existing PORV's and block valves are further being upgraded to include environmentally qualified components and will be installed as soon as the components become available and at an outage of sufficient duration. Presently, component lead times are preventing a 1/1/80 commitment for completion of this upgrading.

The design of the PORV's are such that they rely on nitrogen rather than instrument air. Three of the pressurizer level instrumentation are powered from a vital instrument bus which is fed from a battery/inverter system for reliability. The battery charger can be powered form the diesel generators when the offsite power is not available. The fourth is fed from a vital instrument bus which derives its power from a 480 Motor Control Center (MCC) and a constant voltage transformer. The MCC can be tied to the emergency diesel from the control room.

Section 2.1.2 - Performance Testing for PWR Relief and Safety Valves

The Authority is a member of the Westinghouse PWR Owner's Group. As a member of this group, the Authority is working with Westinghouse, the other PWR owners and the Electric Power Research Institute (EPRI) to develop a program for qualification of relief and code safety valves under expected operating and transient conditions. The program description and schedule will be submitted by the date specified, January 1, 1980, by each utility member when the information is made available.

Section 2.1.3.a - Direct Indication of Power-Operated Relief Valve and Safety Valve Position for PWRs and BWRs

All of the pressurizer power operated relief valves (PORV) and their associated motor-operated block valves (MOV) have positive position indication in the control room. In addition, downstream temperature indication, Pressurizer Relief Tank temperature, level and pressure indication will also indirectly indicate valve position.

While we firmly believe that the existing instrumentation is sufficient for detecting leakage from the code safety relief valves, via indirect valve indication as mentioned above, an acoustic monitoring system for position indication of these valves will be installed. At this time, none of the commercially available equipment has been qualified to safety grade standards; however, testing is going on to support qualification. Present delivery time on off-the-shelf systems has been quoted as up to 16 weeks after receipt of order. We are therefore proceeding to purchase, design and install the acoustic monitoring system by January 1, 1980, but may not be able to meet this date because of equipment long lead time.

Continued operation of the plant pending installation of the acoustic monitoring system for the safety valves is acceptable and could not lead to a TMI type of event for the following reasons:

a) the PORVs and the MOVs have positive position indication.

b) Temperature sensing elements are provided downstream of each safety valve. An additional temperature sensor is located in the common manifold joining the discharge of the PORVs and safety valves. The readouts for this instrumentation are in the Central Control Room.

c) The discharges from the safety valves are piped into the pressurizer relief tank (PRT) which has pressure, temperature and liquid level sensors. The readouts for this instrumentation is located in the Central Control Room.

d) Alarms are provided in the Central Control Room for the following off-normal conditions:

- High safety valve discharge temperature
- High manifold temperature
- High temperature in the PRT
- High pressure in the PRT
- High liquid level in the PRT

e) Throughout the operating history of the plant, the PORVs and safety valves have never been challenged.

With all of the above instrumentation, alarms and positive indication of the PORVs, all of which are control grade and powered from the emergency bus, the operator is able to identify an abnormal condition. By closing the MOVs, the leakage path can be identified and corrective action taken. Existing procedures require that the PRT temperature, pressure and liquid level be logged-in twice a shift.

Section 2.1.3.b - Instrumentation for Detection of Inadequate Core Cooling in PWR's

The Westinghouse PWR Owner's Group is performing calculations associated with the definitions and identification of inadequate core cooling conditions in accordance with NUREG-0578 Section 2.1.9. This work is being reviewed by the Bulletins and Order Task Force of the NRC. The schedule by which this work will be performed is being determined by the Bulletins and Order Task Force. The results of this program will provide guidelines which can be implemented into plant specific procedures and identify existing plant instrumentation which can be utilized in assessing the approach to and existence of inadequate core cooling.

The program as described in the preceding paragraph, will be utilized to provide the basis for any new instrumentation or controls which may be required in assessing the approach to an existence of inadequate core cooling. A description of any new instrumentation or controls, their functional design requirements and a schedule for installation will be provided consistent with the analysis of Section 2.1.9.

A subcooling meter is being designed and components procured. It will consist of selector switches to choose one out of two safety grade pressure sensors, one out of four safety grade temperature sensors or selection of one out of a number of incore thermocouples. The difference in reactor coolant system pressure from saturation will be indicated on a recorder in the Control Room.

However, we expect to complete installation of the subcooling meter with single channel temperature and pressure inputs. The equipment necessary for redundant channel inputs will not be available to support a January 1, 1980 deadline. The subcooling meter will be upgraded to include redundant channel inputs as soon as the equipment is available. To date, we have not received a firm delivery date of this equipment.

In lieu of the above system, a supplemental means utilizing the plant computer is being programmed and will continuously read the subcooling conditions. This system will compute the saturation pressure due to either the average hot leg temperature or the average incore thermocouples. The reactor coolant pressure is then compared to this and the difference can be displayed on the Cathode Ray Tube (CRT) or logged on the computer recorders. Two alarms will annunciate depending on whether the reactor coolant is below the saturation pressure or at a preset band above the saturation pressure.

The above mentioned plant modification will not adversely impact the reactor protection or engineering safeguard system.

Further information on the subcooling meter will be provided by 1/1/80 as requested by the attachment in section 2.1.3.b of the 10/30/79 letter.

A procedure is in effect to recognize inadequate core cooling. It will be revised to include the use of the subcooling meter, as well as other related plant parameters.

Section 2.1.5.a - Dedicated Hydrogen Control Penetrations

Indian Point 3 has redundant hydrogen recombiners located in containment. No requirement for external recombiners is associated with this facility, therefore, no further action is required on this item.

Section 2.1.5.b - Rulemaking to Require Inerting BWR Containments

Not applicable to IP-3.

Section 2.1.5.c - Capability to Install Hydrogen Recombiners at Each Light Water Nuclear Power Plant

A confirmatory review of the existing hydrogen recombiners system is being conducted. The procedures and basis for recombiner operation along with shielding requirements are being reviewed consistent with the requirements of Section 2.1.6.b of NUREG-0578.

The redundant hydrogen recombiners are located in containment. The recombiner control panels and support systems are located external to containment in an area that will remain accessible during the post-accident period. A review of the shielding for the panel area consistent with Reg. Guide 1.4 is being conducted as part of Section 2.1.6.b of this program. This review will be completed by January 1, 1980.

Section 2.1.6.a - Integrity of Systems Outside Containment Likely to Contain Radioactive Materials (Engineered Safety Systems and Auxiliary Systems) for PWR's.

The Authority is presently reviewing lines external to containment that normally transfer nonradioactive liquids and gases that may become contaminated with radioactive material in the event of an accident. Upon completion of this review the Authority will research the existing test program for these lines and develop testing procedures and frequency for those potentially contaminated lines which presently have no testing program. By January 1, 1980, we will provide a summary of the program to reduce leakage. In addition, the incident that occurred at North Anna, Unit #1, will be considered, and any modifications which are deemed necessary will be detailed by 1/1/80, including schedules for completion.

Section 2.1.6.b - Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May be Used in Post-Accident Operations

The following actions have been planned to comply with NUREG-0578:

1. Identification and field verification of reactor coolant recirculation paths, operating and sampling stations under accident and post-accident conditions.

2. A design review is being conducted to identify safety equipment which may be unduly degraded by the radiation fields during post-accident operations of these systems.

3. Determination of radiation fields resulting from circulation of reactor coolant following a release equivalent to that described in Regulatory Guide 1.4.

4. Determination of dose rates based on accident source terms at operating and sampling stations and approaches

which are to be used by plant personnel during accident and post-accident conditions.

5. Identification of modifications necessary for personnel to accomplish essential functions and to upgrade safety equipment, which may be degraded due to an adverse environment.

6. A report summarizing the above will be submitted by January 1, 1980.

Section 2.1.7.a - Auto Initiation of the Auxiliary Feedwater System

A confirmatory review has been conducted of the initiation logic design, operational and administrative procedures of the Auxiliary Feedwater System for compliance with the NRC's clarification letter of October 30, 1979. This review has concluded that the system complies with Section 2.1.7.a. The results of this review are presented below:

1. The AFWS has auto/manual initiation.

2. Testability exists for the initiating signal and circuits.

3. Initiating signals and circuits are powered from emergency buses.

4. Necessary pumps are automatically sequenced on to the emergency buses as part of the normal safety injection scheme. The addition of these loads will not compromise the diesel's generating capacity. The power to the valves are always available via a vital instrument bus.

5. Failure of the automatic circuits will not result in loss of manual initiation capability from the control room.

6. The instrument air supply to necessary valves in the Auxiliary Feedwater Building is backed up by an H_2 bottle system. In addition, the valves can be manually operated.

Based upon this review, the operation of the IP-3 facility does not present an undue hazard to the public health and safety.

Section 2.1.7.b - Auxiliary Feedwater Flow Indication to Steam Generators for PWR's

A confirmatory review of the Auxiliary Feedwater flow indication to steam generators in the control room has been conducted.

Based on the clarification letter issued by the NRC dated 10/30/79, the resultsof this review are as follows.

The auxiliary feedwater flow instrumentation provide local and remote indication which are powered from the vital instrument buses.

Flowmeters are provided in the common auxiliary feedwater connections to the main feedwater line of each steam generator. These indicate flow remotely in the control room and also in the auxiliary feed pump room. Steam generator water level indicators also are located in the control room and auxiliary feed pump room which are powered from the vital instrument buses. The present configuration meets the single failure criteria considering the S.G. level as a backup to the flow indicators in the control room. It is worth noting that loss of flow indication (and flow) to one steam generator will not compromise the heat removal capability from the reactor. Testability exists for the flow and level indications. The instrument error inaccuracy of the flow and level indicator channels are less than 10%.

Based upon the above finding, the operation of the IP-3 facility does not present an undue hazard to the public health and safety.

Section 2.1.8.a - Improved Post Accident Sampling Capability

A review of the shielding requirements necessary for sampling and analysis required under accident conditions without incurring a radiation exposure to any individual in excess of permissible limits is in progress and will be completed in December. Upon completion of this review, procedures necessary to perform sampling and analysis will be prepared and implemented by January 1, 1980.

Section 2.1.8.b - Increased Range of Radiation Monitors

The review addressed in 2.1.8.a. above is also determining accessibility of locations for monitoring plant effluents over the ranges required in this section. Upon completion of this review, procedures necessary to perform this monitoring will be prepared and implemented by January 1, 1980.

Specifications for high range effluent monitors are being prepared. Orders will be placed for delivery and installation of such monitors consistent with a January 1, 1981 schedule date. However, the Authority cannot guarantee this date due to long equipment procurement lead time.

Section 2.1.8.c - Improved In-Plant Iodine Instrumentation Under Accident Conditions

We presently have instrumentation capable of sampling atmospheres for iodine-131 and cart-mountable single channel analyzers to evaluate collected samples. New procedures will be prepared and implemented by January 1, 1980 to use this instrumentation under accident conditions to assess the need for respiratory protection.

Section 2.1.9 - Analysis of Design and Off-Normal Transients and Accidents

Transient and Accident Analysis

Analysis of small break loss-of-coolant accidents, symptoms of inadequate core cooling, required action to restore core cooling, and analysis of transient and accident scenarios including operator actions not previously analyzed, are being performed on a generic basis by the Westinghouse PWR Owners Group. The small break analyses have been completed and are reported in WCAP-9600, which was submitted to the Bulletins and Orders Task Force by the Owners Group on June 29, 1979. The work required to address the other two areas, inadequate core cooling and the transient and accident scenarios, is being performed in conjunction with the Bulletins and Order Task Force. The definition of requirements and schedules for submittal of program results are being established with the B&O task force. The results of these programs will provide the bases for the adequacy of existing instrumentation or define the requirements for additional instrumentation or controls in accordance with Item 2.1.3.b.

In addition to the above-outlined program, the Owners Group is providing pre-test predictive analysis of the LOFT test program in accordance with the schedule established by the Bulletins and Order Task Force.

Containment Pressure Monitor

A review of the existing containment pressure monitoring system design is being conducted to check compliance with Regulatory Guides 1.89 and 1.97. An implementation schedule to meet January 1, 1981 installation dictated by NUREG-0578 will be pursued. Component lead time, equipment and unit availability may impact this schedule date.

Containment Water Level Monitor

A review of the containment water level monitoring system will be conducted to check compliance with the NUREG-0578. Implementation schedule of modifications, if required, will be dictated by component lead time, equipment and unit availability.

Containment Hydrogen Monitor

A review to ascertain the extent of modifications required to provide continuous indication of hydrogen concentration in the control room will be undertaken. An implementation schedule to meet January 1, 1981 installation dictated by NUREG-0578 will be pursued. Component lead time, equipment and unit availability may impact this schedule date.

RCS Venting System

The Authority is presently reviewing a preliminary design from Westinghouse for a reactor coolant head vent system. Upon acceptance, the Authority will proceed with detail design and engineering for installation into the plant system in keeping with NUREG-0578. Installation will be completed based on component lead time, equipment and unit availability. A design review will be completed by 1/1/80.

2.2.1.a - Shift Supervisor Responsibilities

1. The Power Authority commits to having the highest level of corporate management issue and periodically reissue a management directive that emphasizes the primary management responsibility of the shift supervisor for safe operation of the plant under all conditions on his shift and that clearly establishes his command duties.

2. Plant procedures will be revised to assure that the duties, responsibilities and authority of the shift supervisor and control room operators are properly defined to effect the establishment of a definite line of command and clear delineation of the command decision authority of the shift supervisor in the control room relative to other plant management personnel. The senior reactor operator will be designated to assume the control room command function in the absence of the shift supervisor from the control room.

The procedure will include the responsibility and authority of the shift supervisor to maintain the broadest perspective of operational conditions affecting the safety of the plant. However, for certain types of events, it may be more effective for the shift supervisor to leave the control room to obtain first hand knowledge of a critical situation and to ensure that the proper steps are initiated to mitigate the consequences of that situation.

Upon the absence of the shift supervisor from the control room, the transfer of authority with another qualified senior reactor operator will be accomplished verbally. In a time of crisis, it is not always prudent to document the transfer of authority as it will detract from the concentration of the accident.

3. The training program for shift supervisors in the responsibility for safe operation and the management function will be expanded to include the requirements of ANS 3.1 (Draft) section 5.2.1.8.

4. The Power Authority commits to having the administrative duties of the shift supervisor reviewed by the senior officer responsible for plant operations. Administrative functions that detract from or are subordinate to the management responsibility for assuring the safe operation of the plant will be delegated to other operations personnel not on duty in the control room.

<u>2.2.1.b - Shift Technical Advisor</u>

The Authority will provide a shift technical advisor as required, by January 1, 1980.

<u>2.2.1.c - Shift and Relief Turnover Procedure</u>

1. A check list will be provided for the control room operators as required in this item.

2. A checklist or log will be provided for the auxiliary operators but is felt not necessary for technicians. Operations personnel remove equipment from service for work done by technicians or authorize removal, and subsequently restore the equipment to service or verify restoration, following completion of the work.

3. Presently independent verification of selected system alignments is periodically made. However, a more formal system will be established to evaluate the effectiveness of the shift and relief turnover procedure.

Section 2.2.2.a - Control Room Access

Administrative procedures will be written and/or revised to limit control room access and establish clear lines of authority and responsibility consistent with the requirements of Section 2.2.2.a of NUREG-0578.

Section 2.2.2.b - Onsite Technical Support Center

An interim Technical Support Center will be established consistent with the requirements of section 2.2.2.b by January 1, 1980. The Power Authority is pursuing the establishment of an upgraded Technical Support Center in the new Administrative Building by January 1, 1981, but may not be able to meet the January 1, 1981 date for all the Technical Support Center requirements because of long equipment lead times.

Section 2.2.2.c - Onsite Operational Support Center

An Operational Support Center will be established consistent with the requirements of Section 2.2.2.c of NUREG-0578.

Items Covered by Enclosure 7 and 8 to the September 13, 1979 NRC letter

Near Term Emergency Preparedness Improved Implementation

1) Upgrade Emergency Plan

The Emergency Plan for Indian Point No. 3 Nuclear Power Plant meets the requirements of Regulatory Guide 1.101, Revision 1 and the new NRC acceptance criteria for licensee emergency plans. This revised emergency plan was submitted to the NRC on November 9, 1979 for review.

2) Short Term Actions Recommended by Lessons Learned Task Force

Section 2.1.8.a. Improved Post Accident Sampling Capability

A review of the shielding requirements necessary for sampling and analysis required under accident conditions without incurring a radiation exposure to any individual in excess of permissible limits is in progress and will be completed in December. Upon completion of this review, procedures necessary to perform sampling and analysis will be prepared and implemented by January 1, 1980.

Section 2.1.8.b Increased Range of Radiation Monitors

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Specifications for high range effluent monitors are being prepared. Orders will be placed for delivery and installation of such monitors by January 1, 1981.

Section 2.1.8.c Improved In-Plant Iodine Instrumentation Under Accident Conditions

We presently have instrumentation capable of sampling atmospheres for iodine-131 and cart-mountable single channel analyzers to evaluate collected samples. New procedures will be prepared and implemented by January 1, 1980 to use this instrumentation under accident conditions to assess the need for respiratory protection.

3) Emergency Operation Center for Federal, State and Local Officials

The temporary emergency control center for Federal, State and Local Officials exists for the Indian Point 3 Nuclear Power Plant. The Authority plans to review and take necessary steps to meet the long term requirement.

4) Improved Off-Site Monitoring Capability

It is the Authority's position that the Indian Point No. 3 Nuclear Power Plant is already in compliance, but as per Memorandum of Understanding No. 28 with Consolidated Edison Co., additional TLDs and air sampling locations are being installed.

5) Adequacy of State/Local Plans

The New York State Radiological Response Plan is an approved NRC plan. It only has to be upgraded in some areas. The Authority in cooperation with Consolidated Edison Co., has offered the local counties technical expertise to write a model emergency plan that could be made county specific by them. Work on the model plan for the counties has commenced.

6) Conduct of Test Exercises

Section 8.1.2 Drills and Exercises of the revised Indian Point 3 Emergency Plan provides for a joint exercise involving Federal, State and Local response organizations to be conducted once every five years in addition to emergency plan tests already specified in Book 1 of the plan itself.

Section 2.1.4 - Containment Isolation Provisions for PWRs and BWRs

1. The Containment Isolation System at Indian Point 3 satisfies the recommendations of SRP 6.2.4 that there be diversity in the parameters sensed for the initiation of containment isolation. Automatically tripped isolation valves are actuated to the closed position by one of two separate containment isolation signals. The first of these signals (Phase A) isolates all non-essential systems and is derived in conjunction with automatic safety injection initiation based on the following parameters:

   a. Containment Pressure (high)
   b. Low Pressurizer Pressure
   c. Steam Line Differential Pressure
   d. High Steam Line Flow (coincident with low steam line pressure or low Tavg)
   e. Manual Containment Isolation

   The second signal (Phase B) isolates essential systems and is derived in conjunction with automatic containment spray actuation based on containment pressure (high-high). In addition to the Phase A and B signals a high radiation signal inside containment also isolates the containment purge supply and exhaust ducts and the containment pressure relief venting system.

2. To provide compliance with the NUREG position, a review of the classification of essential and non-essential systems will be completed, with the results reported by 1/1/80.

3. In accordance with the FSAR, all normally open non-essential systems are automatically isolated by Phase A containment isolation signal. Any valves which are found not to meet this criteria, based on the review referenced in item 2 above and possible reclassification, will be modified or manually isolated except for the times required to maintain plant operation.

4. A review of the containment isolation system at Indian Point No. 3 indicates that there are a number of valves which automatically reset to the previous position upon reset of containment Phase A isolation. At present, these valves are under operator control via operating procedures to be placed in the closed position prior to resetting Phase A. These valves will be modified to preclude automatic opening on reset based on component lead time, equipment and unit availability. As this review and design has not been completed, the January 1, 1980 date may be difficult to achieve.