Peter Zarakas
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October 17, 1979

Re: Indian Point Unit No. 2
Docket No. 50-247

Director of Nuclear Reactor Regulation
ATTN: Mr. Darrell G. Eisenhut, Acting Director
Division of Operating Reactors
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Eisenhut:

In accordance with your September 13, 1979 letter, the attached document provides our commitments to meet the requirements specified in NUREG-0578.

Should you or your staff have any questions, please contact us.

Very truly yours,

Peter Zarakas
Vice President

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RESPONSE TO NUREG-0578:

"TMI-2 LESSONS LEARNED TASK FORCE STATUS REPORT AND SHORT-TERM RECOMMENDATIONS"

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Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
October, 1979

Section 2.1.1 - Emergency Power Supply Requirements for the Pressurizer Heaters, Power-Operated Relief Valves and Block Valves, and Pressurizer Level Indicators in PWRs

N.R.C. Position on Pressurizer Heater Power Supply

1. The pressurizer heater power supply design shall provide the capability to supply, from either the offsite power source or the emergency power source (when offsite power is not available), a predetermined number of pressurizer heaters and associated controls necessary to establish and maintain natural circulation at hot standby conditions. The required heaters and their controls shall be connected to the emergency buses in a manner that will provide redundant power supply capability.

2. Procedures and training shall be established to make the operator aware of when and how the required pressurizer heaters shall be connected to the emergency buses. If required, the procedures shall identify under what conditions selected emergency loads can be shed from the emergency power source to provide sufficient capacity for the connection of the pressurizer heaters.

3. The time required to accomplish the connection of the preselected pressurizer heater to the emergency buses shall be consistent with the timely initiation and maintenance of natural circulation conditions.

4. Pressurizer heater motive and control power interfaces with the emergency buses shall be accomplished through devices that have been qualified in accordance with safety-grade requirements.

Response to Position No. 1

The existing design of the pressurizer heater power supply meets the requirements of the Commission's position. Four banks of pressurizer heaters are connected to four different 480 volt safeguards switchgear bus sections. The Westinghouse Owners' Group analysis has determined that the minimum requirements to maintain subcooling, in a four loop plant is 150 kw. The existing configuration of heater banks permits the meeting of this power requirement and thus assuring the maintenance of adequate natural circulation.

Response to Position No. 2

Existing procedures provide information to the control room operator on how and when the required pressurizer heaters are to be loaded back on to the emergency buses. The operators are trained in the use of these procedures.

The procedures will be reviewed to determine if any additional specific directions are required. Any procedural modifications that are deemed necessary will be made by January 1, 1980.

Response to Position No. 3

All four pressurizer heater groups receive strip signals from both redundant logic trains on "S.I." or loss of bus voltage. The time required to manually load the heaters back onto the emergency buses is well within the one hour time frame, established by the Westinghouse Owners' Group, to prevent loss of subcooling and potential degradation of natural circulation.

Should offsite power be lost while the plant is in the natural circulation mode, it will take less than one minute to restore the heaters back onto the emergency buses.

Response to Position No. 4

Power supplies to each of the pressurizer heater groups are connected to the 480 volt safeguards buses by safety grade circuit breakers. Control signals are interfaced with instrument bus protection circuitry through isolation amplifiers ("I to I" converters) which are considered part of the protection system in accordance with the criteria of I.E.E.E. 279.

Section 2.1.1 (Cont'd)

N.R.C. Position on Power Supply For Pressurizer Relief and Block Valves and Pressurizer Level Indicators

1. Motive and control components of the power-operated relief valves (PORVs) shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available.

2. Motive and control components associated with the PORV block valves shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available.

3. Motive and control power connections to the emergency buses for the PORVs and their associated block valves shall be through devices that have been qualified in accordance with safety-grade requirements.

4. The pressurizer level indication instrument channels shall be powered from the vital instrument buses. These buses shall have the capability of being supplied from either the offsite power source or the emergency power source when offsite power is not available.

Response to Position No. 1

The existing design of the motive and control power circuitry of the power operated relief valves (PORV) meets the Commission's position. Each PORV receives power from separate and independent inverter fed buses with 125 volt D.C. battery power as a backup.

Response to Position No. 2

The existing design of motive and control power circuitry for the PORV block valves meets the Commission's position. Both motor-operated block valves receive power from the emergency diesel-generator buses.

Response to Position No. 3

The existing design of motive and control power components for the PORVs and their associated block valves meets the Commission's position. The power supplies to each of the PORVs are connected by safety grade circuit breakers. The power supplies to each of the block valves are connected to the safety related motor control centers by safety grade combination starters.

Response to Position No. 4

The existing design of the power supplies for the pressurizer level instrumentation channels meets the Commission's position. Each of the level indication channels is independently powered from different instrument buses. Each of the buses is inverter fed with emergency 125 volt D.C. battery power as a back-up.

Section 2.1.2 - Performance Testing For BWR and PWR Relief and Safety Valves

N.R.C. Position

Pressurized water reactor and boiling water reactor licensees and applicants shall conduct testing to qualify the reactor coolant system relief and safety valves under expected operating conditions for design basis transients and accidents. The licensees and applicants shall determine the expected valve operating conditions through the use of analyses of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70, Revision 2. The single failures applied to these analyses shall be chosen so that the dynamic forces on the safety and relief valves are maximized. Test pressures shall be the highest predicted by conventional safety analysis procedures. Reactor coolant system relief and safety valve qualification shall include qualification of associated control circuitry piping and supports as well as the valves themselves.

Response

An extensive test program to qualify the reactor coolant system relief and safety valves under expected operating conditions for design basis transients and accidents will be undertaken, on a generic basis, by the Electric Power Research Institute (EPRI). The Westinghouse Owners' Group, of which Con Edison is a member, has authorized Westinghouse to develop the parameters necessary to define the test conditions. At present, we do not have detailed information on the test schedule. However, the program is intended to be responsive to the Commission's requirements.

In addition, Con Edison will perform analyses of the relief and safety valve related piping using, where appropriate, information obtained from the EPRI test program.

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

N.R.C. Position

Reactor system relief and safety valves shall be provided with a positive indication in the control room derived from a reliable valve position detection device or a reliable indication of flow in the discharge pipe.

Response

All of the pressurizer power operated relief valves (PORV) and their associated motor-operated block valves (MOV) have positive position indication.

While we firmly believe that the existing Indian Point Unit No. 2 instrumentation is sufficient for detecting leakage from the code safety relief valves, 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. Present delivery time on off-the-shelf systems has been quoted as up to 16 weeks after receipt of order. The delivery time will be further extended to permit qualification testing. We are therefore proposing to install the acoustic monitoring system at the first outage of sufficient duration, after receipt of the equipment, but in no case later than the next refueling outage which is presently scheduled for December, 1980. During the next refueling, we also plan to replace the position indicating switches on the PORVs with units that have been environmentally qualified to the latest I.E.E.E. standards (323, 344 and 382) which have NRC approval.

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

a) The PORVs and 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 are 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 every four hours.

Section 2.1.3.b - Instrumentation for Detection of Inadequate Core Cooling in PWRs and BWRs

N.R.C. Position

1. Licensees shall develop procedures to be used by the operator to recognize inadequate core cooling with currently available instrumentation. The licensee shall provide a description of the existing instrumentation for the operators to use to recognize these conditions. A detailed description of the analyses needed to form the basis for operator training and procedure development shall be provided pursuant to another short-term requirement, "Analysis of Off-Normal Conditions, Including Natural Circulation" (see Section 2.1.9 of this appendix).

   In addition, each PWR shall install a primary coolant saturation meter to provide on-line indication of coolant saturation condition. Operator instructions as to use of this meter shall include consideration that is not be used exclusive of other related plant parameters.

2. Licensees shall provide a description of any additional instrumentation or controls (primary or backup) proposed for the plant to supplement those devices cited in the preceding section giving an unambiguous, easy-to-interpret indication of inadequate core cooling. A description of the functional design requirements for the system shall also be included. A description of the procedures to be used with the proposed equipment, the analysis used in developing these procedures, and a schedule for installing the equipment shall be provided.

Response to Position No. 1

a) The Westinghouse Owners' Group is performing calculations associated with the definition and recognition of inadequte core cooling. Based on this information, procedures to ensure that the operators utilize currently available instrumentation to recognize inadequate core cooling will be instituted by January 1, 1980.

b) A primary coolant saturation meter will be installed in the Central Control Room by January 1, 1980. The meter will provide a display of the margin to saturation in degrees Fahrenheit or PSIG. The input signals for the meter will be derived from safety grade instrumentation. It is anticipated that the saturation meter and its installation will meet safety grade requirements. As of this time the meter is undergoing seismic and environmental qualification testing.

The operators will be instructed that the saturation meter is not to be used exclusively for evaluating reactor coolant system conditions.

Response to Position No. 2

The need for, as well as the development of additional instrumentation to provide indication of inadequate core cooling is expected to be developed, on a generic basis, by the Westinghouse Owners' Group. As soon as the information is available Con Edison will evaluate it. Where such additional instrumentation is deemed necessary, a commitment to install it will be made to the Commission.

Section 2.1.4 - Containment Isolation Provisions for PWRs and BWRs

N.R.C. Position

1. All containment isolation system designs shall comply with the recommendations of SRP 6.2.4; i.e., that there be diversity in the parameters sensed for the initiation of containment isolation.

2. All plants shall give careful reconsideration to the definition of essential and non-essential systems, shall identify each system determined to be essential, shall identify each system determined to be non-essential, shall describe the basis for selection of each essential system, shall modify their containment isolation designs accordingly, and shall report the results of the re-evaluation to the NRC.

3. All non-essential systems shall be automatically isolated by the containment isolation signal.

4. The design of control systems for automatic containment isolation valves shall be such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves. Reopening of containment isolation valves shall require deliberate operator action.

Response to Position No. 1

Indian Point Unit No. 2 complies with the recommendation of SRP 6.2.4 in that there is diversity in the parameters sensed for the initiation of containment isolation.

"Phase A" isolation automatically closes containment isolation valves in "nonessential" process lines penetrating containment. This isolation signal is initiated by high containment pressure, low pressurizer pressure, high differential pressure between steam lines, high steam flow in 2/4 steam lines coincident with low T_{avg} or low steam line pressure, or by manual initiation.

"Phase B" isolation automatically closes containment isolation valves in "essential" process lines penetrating containment. This isolation signal is initiated by high-high containment pressure or manual initiation.

The containment ventilation penetrations (i.e., the containment purge supply and exhaust lines and pressure relief line), which could potentially provide a direct path from the containment atmosphere to the outside environment, receive automatic closure signals on high containment radioactivity in addition to the independent automatic "Phase A" isolation signals they receive.

Response to Position No. 2

As requested, a re-evaluation of the Indian Point Unit No. 2 containment isolation design has been initiated. The preliminary results of this re-evaluation indicate that no change from present system designations (i.e., essential and nonessential) is necessary and no modifications are required.

The Westinghouse Owners' Group is also conducting a generic review of the definition of essential and nonessential systems as they relate to containment isolation designs. When the results of this generic study are available, we will evaluate their applicability to the Indian Point Unit No. 2 design and provide our final report to the N.R.C. at that time.

Response to Position No. 3

The isolation provisions of all non-essential systems in Indian Point Unit No. 2 meet the Commission's requirements. The containment isolation valves in the nonessential process lines incorporate one of the following three isolation provisions:

(1) Receive automatic closure signals on "Phase A" containment isolation,

(2) Maintained locked closed when containment integrity is required, or

(3) Maintained normally closed under administrative control when containment integrity is required.

Response to Position No. 4

In our April 26, 1979 and June 22, 1979 letters to the Commission, Con Edison addressed the adequacy of the Indian Point Unit No. 2 containment isolation design as part of its response to IE Bulletin 79-06A. The evaluation contained in those submittals demonstrated that containment isolation valves would not automatically reopen upon reset of the engineered safeguards initiation (SI) signal.

With regard to this Position on containment isolation signals, in addition to the separation and independence provided between the two redundant containment isolation logic and power trains, three segregated groupings of automatic isolation valves have been provided within each of these two trains. Each grouping in a train is actuated from its own latching master relay. Each master relay in each train has its own associated reset switch. Inadvertent opening of any containment isolation path would require failure of the operator to follow procedures to return the control switches of both redundant valves in that path to their "close" position (had they been open immediately prior to receipt of the isolation signal) and deliberate operator action to activate the corresponding reset switches in both trains for one of the automatic isolation groupings.

The arrangement in which the automatic isolation valves are assigned to each of the three segregated reset groupings within each train reduces the potential for misoperation by the operator. The first grouping is associated with Containment Ventilation Isolation and includes Containment Purge Supply and Exhaust Valves and Containment Pressure Relief Valves. The second grouping is associated with "Phase B" Containment Isolation and includes component cooling water and seal water services for the reactor coolant pumps. The third grouping consists of nonessential service automatic containment isolation valves which receive a "Phase A" containment isolation signal. Segregating the essential equipment service valves (those most likely to be quickly restored by the operator) and the containment ventilation valves (those open to containment atmosphere and therefore having higher release potentials) from the balance of nonessential service valves reduces the overall potential for inadvertent valve openings.

We believe that the above multiple provisions are adequate to prevent the inadvertent transfer of potentially radioactive fluids from containment to the outside environment. Nevertheless, in order to provide further assurance that such transfer could not occur, we are presently conducting a study of all automatic containment isolation valve circuitry to determine if any additional hardware preventive features in specific valve control circuits are desirable. The results of this study will be furnished to the Commission as soon as completed but no later than January 1, 1980. Any proposed improvements to existing circuitry will be completed by January 1, 1981.

Section 2.1.5.a - <u>Dedicated Penetrations for External Recombiners or Post-Accident Purge Systems</u>

N.R.C. Position

Plants using external recombiners or purge systems for post-accident combustible gas control of the containment atmosphere should/provide containment isolation systems for external recombiner or purge systems that are dedicated to that service only that meet the redundancy and single failure requirements of General Design Criteria 54 and 56 of Appendix A to 10 CFR Part 50, and that are sized to satisfy the flow requirements of the recombiner or purge system.

Response

At Indian Point Unit No. 2, post-accident combustible gas control can be effected by use of either the two redundant hydrogen recombiners located inside containment or the post-accident containment venting (purge) system.

Being located inside containment, the hydrogen recombiner system is not, by design, susceptible to the "shared" penetration and flow throttling problems that were experienced at TMI-2. The only penetrations that must be used to support hydrogen recombiner operation are those associated with the oxygen supply to containment atmosphere and the hydrogen supply to the recombiners themselves. These penetrations are dedicated to those services and the containment isolation provisions for each line satisfy the requirements of General Design Criteria 54 and 56 of Appendix A to 10 CFR Part 50.

In addition to the hydrogen recombiner system, a post-accident containment venting (purge) system is provided for combustible gas control. This system is independent of the normal containment purging and pressure relief systems and is equipped with a dedicated post-accident filtering system (i.e., roughing, HEPA, charcoal). The penetration for this system is dedicated to this service, is sized for the flow expected, and satisfies the containment isolation requirements of General Design Criteria 54 and 56 of Appendix A to 10 CFR Part 50.

In conclusion, the present design of Indian Point Unit No. 2 satisfies the Commission's Position for Lessons Learned Recommendation 2.1.5.a and complete implementation is already effected.

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

N.R.C. Position

1. All licensees of light water reactor plants shall have the capability to obtain and install recombiners in their plants within a few days following an accident if containment access is impaired and if such a system is needed for long-term post-accident combustible gas control.

2. The procedures and bases upon which the recombiners would be used on all plants should be the subject of a review by the licensees in considering shielding requirements and personnel exposure limitations as demonstrated to be necessary in the case of TMI-2.

Response to Position No. 1

As discussed in response to Lessons Learned Recommendation 2.1.5.a, hydrogen recombiners are installed inside the containment building by design at Indian Point Unit No. 2. Therefore, the capability for long-term post-accident combustible gas control via hydrogen recombination already exists.

Response to Position No. 2

Plant emergency procedures are presently implemented for use of the hydrogen recombiners during the long-term post-accident period. These procedures require operation of the hydrogen recombiner system when the containment hydrogen concentration is $\geq 3\%$. Certain manual valving and flanging operations are required to establish the oxygen and hydrogen supplies to containment that are necessary to support recombiner operation. We are currently evaluating these required manual operations from the point of view of shielding and personnel exposure limitations as part of the plant shielding review being conducted in response to Lessons Learned Recommendation 2.1.6.b. We will implement any corrective measures deemed necessary by the results of the shielding review for these lines by January 1, 1981.

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

N.R.C. Position

Applicants and licensees shall immediately implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels. This program shall include the following:

1. Immediate Leak Reduction

   a. Implement all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment.

   b. Measure actual leakage rates with system in operation and report them to the NRC.

2. Continuing Leak Reduction

   Establish and implement a program of preventive maintenance to reduce leakage to as-low-as-practical levels. This program shall include periodic integrated leak tests at a frequency not to exceed refueling cycle intervals.

Response to Position Nos. 1 and 2

The original design of Indian Point Unit No. 2 considered potential leakage paths to the environment from systems, outside of containment, that could contain highly radioactive fluids. As an example, the Technical Specifications require that the external portions of the Residual Heat Removal System be periodically tested and, in addition, contain a maximum allowable leak rate provision.

As a further enhancement of this existing philosophy, Con Edison will meet the Commission's position to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels. By January 1, 1980 Con Edison will:

a) Implement leak reduction measures for all systems that could carry radioactive fluid outside of containment.

b) Where possible, measure the actual leakage rates with the systems in operation. All information obtained will be reported to the Commission.

c) Establish and implement a program of preventive maintenance. As required, this program will include periodic integrated leak tests at a frequency not to exceed refueling cycle intervals.

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

N.R.C. Position

With the assumption of a post-accident release of radioactivity equivalent to that described in Regulatory Guides 1.3 and 1.4, (i.e., the equivalent of 50% of the core radioiodine and 100% of the core noble gas inventory are contained in the primary coolant), each licensee shall perform a radiation and shielding design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The design review should identify the location of vital areas and equipment, such as the control room, radwaste control stations, emergency power supplies, motor control centers, and instrument areas, in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during post-accident operations of these systems.

Each licensee shall provide for adequate access to vital areas and protection of safety equipment by design changes, increased permanent or temporary shielding, or post-accident procedural controls. The design review shall determine which types of corrective actions are needed for vital areas throughout the facility.

Response

We are prepared to implement this recommendation in three phases. The first phase, effort on which has already begun, involves the determination of the nature and scope of the problem, if it exists, at Indian Point Unit No. 2. Studies already underway include radiation field level calculations based on the Lessons Learned source term criteria, location of piping capable of containing large radiation sources, identification of adjacent vital equipment and the potential for radiation related dysfunction, and review of present procedures to determine post-accident access requirements. This review will be completed by January 1, 1980.

Second phase activities involve the determination of realistic solutions to any problems found in the first phase effort. Since there is an obvious precedence relationship between the phase one results and scoping of the phase two effort, a reasonable schedule for phase two completion must await the phase one results.

Phase three involves actual implementation of recommended solutions. The nature and scope of these solutions is presently unknown. We will attempt to meet the required implementation date, however if equipment changes are required, it may not be possible to complete all of them by January 1, 1981.

Section 2.1.7.a - Automatic Initiation of the Auxiliary Feedwater System for PWRs

N.R.C. Position

1. The following requirements shall be implemented in the short term:

   a. The design shall provide for the automatic initiation of the auxiliary feedwater system.

   b. The automatic initiation signals and circuits shall be designed so that a single failure will not result in the loss of auxiliary feedwater system function.

   c. Testability of the initiating signals and circuits shall be a feature of the design.

   d. The initiating signals and circuits shall be powered from the emergency buses.

   e. Manual capability to initiate the auxiliary feedwater system from the control room shall be retained and shall be implemented so that a single failure in the manual circuits will not result in the loss of system function.

   f. The a-c motor-driven pumps and valves in the auxiliary feedwater system shall be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.

   g. The automatic initiating signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFWS from the control room.

2. In the long term, the automatic initiation signals and circuits shall be upgraded in accordance with safety-grade requirements.

Response to Position No. 1

1.a The Auxiliary Feedwater System meets the requirements of this position, since automatic initiation is an existing feature of the design.

1.b The Indian Point Unit No. 2 design provides considerable flexibility, with no loss of function from any of the three (3) auxiliary feed water pumps for many single failures which can be postulated within the actuation system. In addition, due to a common circuit dependency between a portion of the Auxiliary Feedwater System actuation logic and the "fail safe" Auxiliary Boiler Feed Pump 22 Auto-Start valve, any failure or loss of this circuit will start Auxiliary Boiler Feed Pump No. 22. We feel that this improves the overall operational reliability of the AFW system by making many potential failures in the logic system a mode of auto-start for the steam driven Auxiliary Boiler Feed Pump.

The existing actuation system design is acceptable to meet control grade requirements since:

1) The single failure criteria for "SI" actuation of the motor driven Auxiliary Boiler Feed Pumps is met and any single failures which could be postulated for the "Unit Trip Plus Loss of Offsite" start signal should only defeat one (1) pump train.
2) For any single failures including those which could defeat the "Main Boiler Feed Pump Trip" and "Steam Generator Low-Low Level" auto-start signals to the Auxiliary Boiler Feed Pumps, manual initiation could not be prevented for more than one of three trains and the operator has approximately 30 minutes to act to initiate AFW manually.
3) As part of our program for improving safe shutdown capability at Indian Point Unit No. 2, a completely

separate and independent power supply is being provided and will be available to motor driven Auxiliary Feedwater Pump 21 through a manual transfer switch. This installation is scheduled to be completed by the end of the fourth refueling presently planned to begin in December, 1980.

A study is currently underway to develop a design to upgrade the automatic initiating signals and circuits of the Auxiliary Feedwater System to full safety grade requirements. We intend to complete any modifications required by January 1, 1981. The final design will include considerations of recommendations in the plant specific evaluation of the Indian Point Unit No. 2 Auxiliary Feedwater System which the NRC Bulletin & Orders Task Force is in the process of completing.

1.c We will meet the requirements of this position. We presently have test procedures which demonstrate operability of pumps and valves within the Auxiliary Feedwater System. We plan to expand these procedures to include initiating signals and circuits. This program will be developed by January 1, 1980.

1.d The design meets the requirements of this position. Initiating signals and circuitry are all powered from emergency buses.

1.e Our present design accomplishes this requirement since failure of any of the manual control circuitry identified in the response to NRC position 1.g will only affect one of the three Auxiliary Feedwater Pumps.

1.f The motor driven pumps are included in the automatic actuation of the emergency buses. The Indian Point Unit No. 2 design meets the intent of control grade implementation of this requirement, since those valves which are not powered from the emergency buses are designed to "fail" into their required safe shutdown positions on loss of power. The planned safety grade upgrading of the Auxiliary Feedwater System will include reconnection of these valves to the emergency bus. This modification will be made by January 1, 1981.

1.g Our existing design complies with this requirement since:

1. Auxiliary Boiler Feed Pump 22 is provided with manual control switches in the Central Control Room and locally at the pump. Either of these control switches can directly de-energize the auto-start solenoid valve (steam supply) to start Auxiliary Boiler Feed Pump 22.

   In the extremely unlikely event that the "fail safe" designed pneumatic operator does not open the valve, it could be "jacked open" mechanically at the pump room.

2. Both motor driven auxiliary boiler feed pumps are provided with capability for manual control from two separate locations. Each pump is provided

with a control switch in the Central Control Room. A set of "stop-start" buttons is provided locally at each pump. A "local/remote" switch which must be operated to arm the local push buttons also disconnects all automatic circuitry from the pump control circuit when it is operated to the "local" position.

Response to Position No. 2

As indicated in our response to position 1.b, we will upgrade the automatic initiating signals and circuits of the Auxiliary Feedwater System to full safety grade by the end of the fourth refueling outage which is presently scheduled for December, 1980.

Section 2.1.7.b - Auxiliary Feedwater Flow Indication to Steam Generators for PWRs

N.R.C. Position

The following requirements shall be implemented:

1. Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.

2. The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements of the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9.

Response

The existing auxiliary feedwater flow indication system, which measures flow rate to each steam generator, will be upgraded to meet safety grade requirements. The system will have four qualified transmitters (one for each steam generator), and four flow indicators in the Central Control Room. Power for the system will be provided from the emergency buses in accordance with the Commission's criteria.

All required wiring and control room instrumentation will be installed prior to January 1, 1980. The flow transmitters will also be available prior to January 1, 1980 and will be installed at the first outage of sufficient duration.

Section 2.1.8.a - Improved Post-Accident Sampling Capability

N.R.C. Position

A design and operational review of the reactor coolant and containment atmosphere sampling systems shall be performed to determine the capability of personnel to promptly obtain (less than 1 hour) a sample under accident conditions without incurring a radiation exposure to any individual in excess of 3 and 18 3/4 Rems to the whole body or extremities, respectively. Accident conditions should assume a Regulatory Guide 1.3 or 1.4 release of fission products. If the review indicates that personnel could not promptly and safely obtain the samples, additional design features or shielding should be provided to meet the criteria.

A design and operational review of the radiological spectrum analysis facilities shall be performed to determine the capability to promptly quantify (less than 2 hours) certain radioisotopes that are indicators of the degree of core damage. Such radionuclides are noble gases (which indicate cladding failure), iodines and cesiums (which indicate high fuel temperatures), and non-volatile isotopes (which indicate fuel melting). The initial reactor coolant spectrum should correspond to a Regulatory Guide 1.3 or 1.4 release. The review should also consider the effects of direct radiation from piping and components in the auxiliary building and possible contamination and direct radiation from airborne effluents. If the review indicates that the analyses required cannot be performed in a prompt manner with existing equipment, then design modifications or equipment procurement shall be undertaken to meet the criteria.

In addition to the radiological analyses, certain chemical analyses are necessary for monitoring reactor conditions. Procedures shall be provided to perform boron and chloride chemical analyses assuming a highly radioactive initial sample (Regulatory Guide 1.3 or 1.4 source term). Both analyses shall be capable of being completed promptly, i.e., the boron sample analysis within an hour and the chloride sample analysis within a shift.

Response

Present plant design includes the capability of sampling reactor coolant, the discharge of the recirculation and residual heat removal pumps, and the post accident containment atmosphere. As part of the design review being conducted to satisfy Lessons Learned recommendation 2.1.6.b., Con Edison will conduct a design and operational review of the reactor coolant and containment atmosphere sampling systems to determine the ability of personnel to promptly obtain a sample under accident conditions without incurring excessive radiation exposures. Additional design features or necessary modifications will be provided (if necessary) to assure that personnel can promptly and safely obtain the necessary samples.

Additionally, the Westinghouse Owners' Group is conducting a study of onsite radiological analysis to determine the capability to promptly quantify certain radioisotopes which are indicative of core damage. If this review indicates that the analyses required cannot be performed in a prompt manner with existing equipment, appropriate corrective action will be proposed to the Commission.

In accordance with the proposed program for implementation of the Lessons Learned Task Force Short Term Recommendations, we expect that the review, procedure preparation, and description of proposed modifications will be completed by January 1, 1980 and any required modifications completed by January 1, 1981.

Section 2.1.8.b - Increased Range of Radiation Monitors

N.R.C. Position

1. Noble gas effluent monitors shall be installed with an extended range designed to function during accident conditions as well as during normal operating conditions; multiple monitors are considered to be necessary to cover the ranges of interest.

    a. Noble gas effluent monitors with an upper range capacity of 10^5 μCi/cc (Xe-133) are considered to be practical and should be installed in all operating plants.

    b. Noble gas effluent monitoring shall be provided for the total range of concentration extending from normal conditions (ALARA) concentrations to a maximum of 10^5 μCi/cc (Xe-133). Multiple monitors are considered to be necessary to cover the ranges of interest. The range capacity of individual monitors should overlap by a factor of ten.

2. Since iodine gaseous effluent monitors for the accident condition are not considered to be practical at this time, capability for effluent monitoring of radioiodines for the accident condition shall be provided with sampling conducted by adsorption on charcoal or other media, followed by onsite laboratory analysis.

3. In-containment radiation level monitors with a maximum range of 10^8 rad/hr shall be installed. A minimum of two such monitors that are physically separated shall be provided. Monitors shall be designed and qualified to function in an accident environment.

Response to Position No. 1

Additional extended-range noble gas effluent monitors designed to function during both accident and normal operating conditions will be installed at Indian Point Unit No. 2. This monitoring capability will provide a range extending from the minimum necessary for normal operating conditions (10^{-7} μCi/cc) to a maximum of 10^5 μCi/cc. (Xe-133). Sufficient range overlap of individual monitors will be provided to assure adequate release categorization capability over the ranges of interest. The necessary plant modifications to meet the Commission's requirements will be completed by January 1, 1981.

Response to Position No. 2

The capability for effluent monitoring of radioiodines, under accident conditions, by sampling using adsorption on charcoal or other suitable media as well as onsite facilities to analyze the samples will be established by January 1, 1981.

Response to Position No. 3

Con Edison will install at least two physically separated high range containment radiation monitors with sufficient range to measure photon radiation fields up to 10^7R/hr. These monitors will be provided with continuous indication in the Central Control Room and will be environmentally qualified.

Information obtained to date indicates that the monitors require coaxial electrical penetrations with a rating of between 600 and 1000V. A study has been started to determine if any of the existing containment electrical penetrations can be used.

While we will make every attempt to have the monitors operational by January 1, 1981, this schedule may not be possible if new containment electrical penetrations are required. Should a longer implementation time be required, the Commission will be notified as soon as a realistic schedule has been established.

Section 2.1.8.c - Improved In-Plant Iodine Instrumentation

N.R.C. Position

Each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident.

Response

To assure the accuracy of airborne iodine concentration measurements within Indian Point Unit No. 2, Con Edison will review its existing equipment and associated training and procedures for accurately determining the airborne iodine concentration in the plant under accident conditions.

Any equipment modifications and procedural changes necessary to assure this capability will be implemented by January 1, 1980.

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

N.R.C. Position

Analyses, procedures, and training addressing the following are required:

1. Small break loss-of-coolant accidents;

2. Inadequate core cooling; and

3. Transients and accidents.

Some analysis requirements for small breaks have already been specified by the Bulletins and Orders Task Force. These should be completed. In addition, pretest calculations of some of the Loss of Fluid Test (LOFT) small break tests (scheduled to start in September 1979) shall be performed as means to verify the analyses performed in support of the small break emergency procedures and in support of an eventual long term verification of compliance with Appendix K of 10 CFR Part 50.

In the analysis of inadequate core cooling, the following conditions shall be analyzed using realistic (best-estimate) methods:

1. Low reactor coolant system inventory (two examples will be required - LOCA with forced flow, LOCA without forced flow).

2. Loss of natural circulation (due to loss of heat sink).

These calculations shall include the period of time during which inadequate core cooling is approached as well as the period of time during which inadequate core cooling exists. The calculations shall be carried out in real time far enough that all important phenomena and instrument indications are included. Each case should then be repeated taking credit for correct operator action. These additional cases will provide the basis for developing appropriate emergency procedures. These calculations should also provide the analytical basis for the design of any additional instrumentation needed to provide operators with an unambiguous indication of vessel water level and core cooling adequacy (see Section 2.1.3.b in this appendix).

The analyses of transients and accidents shall include the design basis events specified in Section 15 of each FSAR. The analyses shall include a single active failure for each system called upon to function for a particular event. Consequential failures shall also be considered. Failures of the operators to perform required control manipulations shall be given consideration for permutations of the analyses. Operator actions that could cause the complete loss of function of a safety system shall also be considered. At present, these analyses need not address passive failures or multiple system failures in the short term. In the recent analysis of small break LOCAs, complete loss of auxiliary feedwater may be added to the failures being considered in the analysis of transients and accidents if it is concluded that more is needed in operator training beyond the short-term actions to upgrade auxiliary feedwater system reliability. Similarly, in the long term, multiple failures and passive failures may be considered depending in part on staff review of the results of the short-term analyses.

The transient and accident analyses shall include event tree analyses, which are supplemented by computer calculations for those cases in which the system response to operator actions is unclear or these calculations could be used to provide important quantitative information not available from an event tree. For example, failure to initiate high-pressure injection could lead to core uncovery for some transients, and a computer calculation could provide information on the amount of time available for corrective action. Reactor simulators may provide some information in defining the event trees and would be useful in studying the information available to the operators. The transient and accident analyses are to be performed for the purpose of identifying appropriate and inappropriate operator actions relating to important safety considerations such as natural circulation, prevention of core uncovery, and prevention of more serious accidents.

The information derived from the preceding analyses shall be included in the plant emergency procedures and operator training. It is expected that analyses performed by the NSSS vendors will be put in the form of emergency procedure guidelines and that the changes in the procedures will be implemented by each licensee or applicant.

In addition to the analyses performed by the reactor vendors, analyses of selected transients should be performed by the NRC Office of Research, using the best available computer codes, to provide the basis for comparisons with the analytical methods being used by the reactor vendors. These comparisons together with comparisons to data, including LOFT small break test data, will constitute the short-term verification effort to assure the adequacy of the analytical methods being used to generate emergency procedures.

Response

Analyses of small break loss-of-coolant accidents, symptoms of inadequate core cooling and required actions 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 Owners' Group. The small break analysis has 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. Incorporated in that report were guidelines that were developed as a result of the small break analyses. These guidelines have been reviewed with the B&O Task Force and will be presented to the Owners' Group utility representatives at a seminar held on October 16-19, 1979. Based on the information from this seminar, Con Edison will develop plant specific procedures and train personnel, on the new procedures, by January 1, 1980.

The analyses required to address the other two areas -- inadequate core cooling and other transient and accident scenarios -- are being performed in conjunction with the Bulletins and Orders Task Force, including establishment of information requirements to meet the duties specified in Enclosure 6 of the Commission's September 13, 1979 letter. Analyses related to the definition of inadequate core cooling and guidelines for recognizing the symptoms of inadequate core cooling based on existing plant instrumentation and recovery from such a condition will be completed by October 31, 1979. Further work to better define the approaches to inadequate core cooling and recovery operations may be required and will be performed later. It is intended that the guidelines provided by October 31, 1979, will be incorporated into plant procedures and training accomplished by the required date of January 1, 1980.

Analyses related to other transients and accidents contained in Chapter 14 of the Indian Point Unit No. 2 FSAR will be provided by the required date of January 1, 1980.

The Owners' Group is also providing pretest prediction analysis of the LOFT L3-1 nuclear small break experiment. This analysis will be submitted by the required date of November 15, 1979, in accordance with the schedule established by the B&O Task Force.

ACRS Item - <u>Instrumentation to Monitor Containment Conditions During the Course of an Accident</u>

N.R.C. Position

Consistent with satisfying the requirements set forth in General Design Criterion 13 to provide the capability in the control room to ascertain containment conditions during the course of an accident, the following requirements shall be implemented:

(1) A continuous indication of containment pressure shall be provided in the control room. Measurement and indication capability shall include three times the design pressure of the containment for concrete, four times the design pressure for steel, and minus five psig for all containments.

(2) A continuous indication of hydrogen concentration in the containment atmosphere shall be provided in the control room. Measurement capability shall be provided over the range of 0 to 10% hydrogen concentration under both positive and negative ambient pressure.

(3) A continuous indication of containment water level shall be provided in the control room for all plants. A narrow range instrument shall be provided for PWRs and cover the range from the bottom to the top of the containment sump. Also for PWRs, a wide range instrument shall be provided and cover the range from the bottom of the containment to the elevation equivalent to a 500,000 gallon capacity.

The containment pressure, hydrogen concentration and wide range containment water level measurements shall meet the design and qualification provisions of Regulatory Guide 1.97, including qualification, redundancy, and testability. The narrow range containment water level measurement instrumentation shall be qualified to meet the requirements of Regulatory Guide 1.89 and shall be capable of being periodically tested.

Response to Position No. 1

Con Edison will install new instrumentation, by January 1, 1981, which will meet safety grade requirements and be capable of measuring containment pressures from -5 PSIG to three times the design accident pressure. The system will provide a continuous readout in the Central Control Room.

Response to Position No. 2

Con Edison will install hydrogen monitoring systems, by January 1, 1981, which will meet safety grade requirements and be capable of measuring the hydrogen concentration, inside of containment over a range of 0 to 10%. The system will provide a continuous readout in the Central Control Room.

Response to Position No. 3

Con Edison will install two new wide-range liquid level monitoring systems, by January 1, 1981 which will meet safety grade requirements and be capable of measuring water levels inside of containment from the bottom of the sump to the 500,000 gallon level. The system will provide a continuous readout in the Central Control Room.

The existing two water level monitoring systems (containment sump and recirculation sump) will be retained for narrow-range use and to provide redundancy for the new wide range system. The existing systems consist of hermetically sealed magnetic switches in stainless steel housings. The instrumentation is designed to be testable and for submerged service in 295°F borated water at a pressure of 69 psig. Since instruments of this design have been subjected to considerable actual service in applications more severe than post LOCA design conditions, environmental testing for these instruments is not required. Both systems provide level indication in the Central Control Room through a series of lights designating incremental water levels.

Staff Item - Installation of Remotely Operated High Point Vents In The Reactor Coolant System

N.R.C. Position

Each applicant and licensee shall install reactor coolant system and reactor vessel head high point vents remotely operated from the control room. Since these vents form a part of the reactor coolant pressure boundary, the design of the vents shall conform to the requirements of Appendix A to 10 CFR Part 50 General Design Criteria. In particular, these vents shall be safety grade, and shall satisfy the single failure criterion and the requirements of IEEE-279 in order to ensure a low probability of inadvertent actuation.

Each applicant and licensee shall provide the following information concerning the design and operation of these high point vents:

1. A description of the construction, location, size, and power supply for the vents along with results of analyses of loss-of-coolant accidents initiated by a break in the vent pipe. The results of the analyses should be demonstrated to be acceptable in accordance with the acceptance criteria of 10 CFR 50.46.

2. Analyses demonstrating that the direct venting of noncondensable gases with perhaps high hydrogen concentrations does not result in violation of combustible gas concentration limits in containment as described in 10 CFR Part 50.44, Regulatory Guide 1.7 (Rev. 1), and Standard Review Plan Section 6.2.5.

3. Procedural guidelines for the operators' use of the vents. The information available to the operator for initiating or terminating vent usage shall be discussed.

Response

Con Edison is currently designing a reactor vessel head high point venting system which will be remotely operated from the Central Control Room. The design and supporting analyses will be provided for Commission review by January 1, 1980. Installation of the system will be during the next refueling outage which is presently scheduled for December, 1980.

The design criteria presented at the commission's October 11, 1979 Topical Meeting in Bethesda, Md., established the acceptability (for Westinghouse reactors) of using a single reactor head vent having at least one motor-operated, remotely controlled valve with flow indication and a second vent on the pressurizer. It was further stated that the pressurizer power operated relief valves (PORV) were acceptable as the second vent provided the Pressurizer Relief Tank (PRT) gas space was inerted with a N_2 blanket gas.

Since the PRT in Indian Point Unit No. 2 is normally operated with a N_2 atmosphere and is provided with remote gas sampling capability, the pressurizer PORVs will be used as the ''second'' vent in the head vent system. As presently envisioned, the ''first'' vent (on the head) will be controlled by two motor operated valves, in series, on separate channels, both normally closed. A qualitative type flow sensor will be provided upstream of the two valves.

Additional reactor coolant system high point vents were evaluated and considered to be unnecessary, impractical and to decrease the margin of safety. As an example, to vent the highest points, which are at the top of the steam generator tube U-bends, would require thousands of penetrations in the reactor coolant system. Such a configuration would significantly increase the probability of a small break LOCA.

Section 2.2.1.a - Shift Supervisor's Responsibilities

N.R.C. Position

1. The highest level of corporate management of each licensee shall 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 shall be reviewed 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. Particular emphasis shall be placed on the following:

    a. The responsibility and authority of the shift supervisor shall be to maintain the broadest perspective of operational conditions affecting the safety of the plant as a matter of highest priority at all times when on duty in the control room. The idea shall be reinforced that the shift supervisor should not become totally involved in any single operation in times of emergency when multiple operations are required in the control room.

    b. The shift supervisor, until properly relieved, shall remain in the control room at all times during accident situations to direct the activities of control room operators. Persons authorized to relieve the shift supervisor shall be specified.

    c. If the shift supervisor is temporarily absent from the control room during routine operations, a lead control room operator shall be designated to assume the control room command function. These temporary duties, responsibilities, and authority shall be clearly specified.

3. Training programs for shift supervisors shall emphasize and reinforce the responsibility for safe operation and the management function the shift supervisor is to provide for assuring safety.

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

Response to Position No. 1

Con Edison will institute a corporate policy by January 1, 1980, whereby the Vice President, Power Generation will issue and reissue annually a management directive that emphasizes the primary management responsibility of the shift supervisor for safe operation of the plant and that clearly establishes his command duties. Copies of the directive will be issued to shift and plant management personnel.

Response to Position No. 2

Plant procedures will be reviewed and modified, if required, by January 1, 1980 to assure that the duties, responsibilities and authority of the shift supervisor and control room operators are properly defined to effect establishment of definite lines of command and command decision authority of the shift supervisor in the control room relative to other plant management personnel.

Response to Position No. 3

The training program for shift supervisors will be reviewed and modified, if required, by January 1, 1980 to emphasize and reinforce the responsibility for safe operation and the management function the shift supervisor is to provide for assuring safety.

Response to Position No. 4

Con Edison will institute a corporate policy by January 1, 1980, whereby the Vice President, Power generation will review and annually rereview the administrative duties of the shift supervisor. Administrative functions that would prevent the shift supervisor from performing his management responsibility for assuring the safe operation of the plant will be delegated to other operations personnel not on duty in the control room.

Section 2.2.1.b - Shift Technical Advisor

N.R.C. Position

Each licensee shall provide an on-shift technical advisor to the shift supervisor. The shift technical advisor may serve more than one unit at a multi-unit site if qualified to perform the advisor function for the various units.

The shift technical advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline and have received specific training in the response and analysis of the plant for transients and accidents. The shift technical advisor shall also receive training in plant design and layout, including the capabilities of instrumentation and controls in the control room. The licensee shall assign normal duties to the shift technical advisors that pertain to the engineering aspects of assuring safe operations of the plant, including the review and evaluation of operating experience.

Response

Con Edison expects to meet the requirement for Shift Technical Advisor with personnel meeting the educational and experience requirements outlined in NUREG-0578 by January 1, 1980 and complete training by January 1, 1981. During off-normal reactor plant conditions, the Shift Technical Advisor will report to the Control Room and serve the Watch Supervisor in an advisory capacity.

Routine duties and assignments of the Shift Technical Advisor will include matters involving engineering evaluation of day-to-day plant operations from a safety point of view.

Section 2.2.1.c - Shift and Relief Turnover Procedures

N.R.C. Position

The licensees shall review and revise as necessary the plant procedure for shift and relief turnover to assure the following:

1. A checklist shall be provided for the oncoming and offgoing control room operators and the oncoming shift supervisor to complete and sign. The following items, as a minimum, shall be included in the checklist:

    a. Assurance that critical plant parameters are within allowable limits (parameters and allowable limits shall be listed on the checklist).

    b. Assurance of the availability and proper alignment of all systems essential to the prevention and mitigation of operational transients and accidents by a check of the control console (what to check and criteria for acceptable status shall be included on the checklist);

    c. Identification of systems and components that are in a degraded mode of operation permitted by the Technical Specifications. For such systems and components, the length of time in the degraded mode shall be compared with the Technical Specifications action statement (this shall be recorded as a separate entry on the checklist).

2. Checklists or logs shall be provided for completion by the offgoing and oncoming auxiliary operators and technicians. Such checklists or logs shall include any equipment under maintenance or test that by themselves could degrade a system critical to the prevention and mitigation of operational transients and accidents or initiate operational transients (what to check and criteria for acceptable status shall be included on the checklist); and

3. A system shall be established to evaluate the effectiveness of the shift and relief turnover procedure (for example, periodic independent verification of system alignments).

Response to Position No. 1

Plant procedures will be reviewed and modified, if required, by January 1, 1980 to include the use of checklist(s) for oncoming and offgoing control room operators and the oncoming shift supervisor to complete and sign. The checklists will contain sufficient information to detail the plant status and operational conditions.

Response to Position No. 2

Plant procedures will be reviewed and modified, if required, by January 1, 1980 to include the use of checklists or logs for shift supervisors. The checklists or logs will note any equipment under maintenance or test that by themselves could degrade a system critical to the prevention and mitigation, as well as initiating, operational transients and accidents.

Response to Position No. 3

A system will be established, by January 1, 1980, to evaluate the effectiveness of the shift and relief turnover procedure(s).

Section 2.2.2.a - Control Room Access

N.R.C. Position

The licensee shall make provisions for limiting access to the control room to those individuals responsible for the direct operation of the nuclear power plant (e.g., operations supervisor, shift supervisor, and control room operators), to technical advisors who may be requested or required to support the operation, and to predesignated NRC personnel. Provisions shall include the following:

1. Develop and implement an administrative procedure that establishes the authority and responsibility of the person in charge of the control room to limit access.

2. Develop and implement procedures that establish a clear line of authority and responsibility in the control room in the event of an emergency. The line of succession for the person in charge of the control room shall be established and limited to persons possessing a current senior reactor operator's license. The plan shall clearly define the lines of communication and authority for plant management personnel not in direct command of operations, including those who report to stations outside of the control room.

Response

Procedures for limiting control room access and establishing a clear line of authority and responsibility in the control room, in the event of an emergency, will be implemented by January 1, 1980.

Section 2.2.2.b - Onsite Technical Support Center

N.R.C. Position

Each operating nuclear power plant shall maintain an onsite technical support center separate from and in close proximity to the control room that has the capability to display and transmit plant status to those individuals who are knowledgeable of and responsible for engineering and management support of reactor operations in the event of an accident. The center shall be habitable to the same degree as the control room for postulated accident conditions. The licensee shall revise his emergency plans as necessary to incorporate the role and location of the technical support center.

Records that pertain to the as-built conditions and layout of structures, systems and components shall be stored and filed at the site and accessible to the technical support center under emergency conditions. Examples of such records include system descriptions, general arrangement drawings, piping and instrument diagrams, piping system isometrics, electrical schematics, wire and cable lists, and single line electrical diagrams. It is not the intent that all records described in ANSI N45.2.9-1974 be stored and filed at the site and accessible to the technical support center under emergency conditions; however, as stated in that standard, storage systems shall provide for accurate retrieval of all pertinent information without undue delay.

Response

Con Edison will establish an onsite Technical Support Center in two phases. The first phase, which will be completed by January 1, 1980, will have the following provisions:

- Space to accomodate at least 25 people.
- Dedicated telephone communications to the Central Control Room, NRC Operations Center and the Emergency Operations Center.
- Data transmission from the Central Control Room using either closed circuit television or a telephone talker.
- Ready access to as built plant drawings, electrical schematics, wire and cable lists and single line electrical diagrams.

The second phase will consist of upgrading the center to meet the commission's requirements for habitability, instrumentation emergency power and data acquisition, display, retention and transmission. The plans for the long range upgrading will be completed and submitted to the Commission by January 1, 1980.

Section 2.2.2.c - Onsite Operational Support Center

N.R.C. Position

An area to be designated as the onsite operational support center shall be established. It shall be separate from the control room and shall be the place to which the operations support personnel will report in an emergency situation. Communications with the control room shall be provided. The emergency plan shall be revised to reflect the existence of the center and to establish the methods and lines of communication and management.

Response

An onsite operational support center will be established by January 1, 1980. Con Edison's Emergency Plan will be revised to reflect the existence of the center and to establish the methods and lines of communication and management.

Staff Item - Near Term Requirements For Improving Emergency Preparedness

N.R.C. Position

Our near term requirements in this effort are as follows:

(1) Upgrade licensee emergency plans to satisfy Regulatory Guide 1.101, with special attention to the development of uniform action level criteria based on plant parameters.

(2) Assure the implementation of the related recommendations of the Lessons Learned Task Force involving instrumentation to follow the course of an accident and relate the information provided by this instrumentation to the emergency plan action levels. This will include instrumentation for post-accident sampling, high range radioactivity monitors, and improved in-plant radioiodine instrumentation. The implementation of the Lessons Learned Task Force's recommendations on instrumentation for detection of inadequate core cooling will also be factored into the emergency plan action level criteria.

(3) Determine that an emergency operations center for Federal, State and local personnel has been established with suitable communications to the plant, and that upgrading of the facility in accordance with the Lessons Learned Task Force's recommendation for an in-plant technical support center is underway.

(4) Assure that improved licensee offsite monitoring capabilities (including additional thermoluminescent dosimeters or the equivalent) have been provided for all sites.

(5) Assess the relationship of State/local plans to the licensees' and Federal plans so as to assure the capability to take appropriate emergency actions. Assure that this capability will be extended to a distance of ten miles. This item will be performed in conjunction with the Office of State Programs and the Office of Inspection and Enforcement.

(6) Require test exercises of approved emergency plans (Federal, State, local and licensees), review plans for such exercises, and participate in a limited number of joint exercises. Tests of licensee plans will be required to be conducted as soon as practical for all facilities and before reactor startup for new licensees. Exercises of State plans will be performed in conjunction with the concurrence reviews of the Office of State Programs. As

a preliminary planning bases, assume that joint test exercises involving Federal, State, local and licensees will be conducted at the rate of about ten per year, which would result in all sites being exercised once each five years. Revised planning guidance may result from the ongoing rulemaking.

Response to Position No. 1

An emergency plan for Indian Point Units Nos. 1 and 2 was prepared in accordance with Reg Guide 1.101 and submitted to the Commission on April 2, 1979. This document will be reviewed and, if required, upgraded by mid-1980 to include a uniform action level criteria based on plant parameters.

Response to Position No. 2

Action levels relating to plant parameters based on existing instrumentation is already incorporated into the emergency plan. As additional instrumentation is installed, the action level criteria will be amended to include the new parameters.

Response to Position No. 3

The State and each county have their own local emergency operations centers with communication links to Indian Point. Representatives from Federal, State and local agencies can be accommodated in Con Edison's Emergency Control Center.

Communications between our Emergency Control Center and the proposed Technical Support Center will be established by January 1, 1980 and upgraded, as required, when the Technical Support Center reaches full implementation status.

Response to Position No. 4

Con Edison's offsite monitoring capabilities, including the use of thermoluminescent dosimeters, will be reevaluated and any modifications, if required, will be implemented by mid 1980.

Response to Position No. 5

Con Edison has and will continue to cooperate with State and local governmental representatives to assure that the proper relationships exist between all of the emergency plans so that appropriate emergency action can be taken.

Response to Position No. 6

Con Edison's Emergency Plan is already being exercised on an annual basis. To assure the adequacy and integration of all emergency plans we will:

a) Provide assistance, if requested, to the state in preparing for a test of their Plan.

b) Participate in joint test exercises with the Federal, State and local governments at the rate of approximately once every five years.