ATTACHMENT I

TECHNICAL SPECIFICATION CHANGES

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket 50-247
August, 1996

3.3 ENGINEERED SAFETY FEATURES

Applicability

Applies to the operating status of the Engineered Safety Features.

Objective

To define those limiting conditions for operation that are necessary (1) to remove decay heat from the core in emergency or normal shutdown situations, (2) to remove heat from containment in normal operating and emergency situations, (3) to remove airborne iodine from the containment atmosphere following a Design Basis Accident, (4) to minimize containment leakage to the environment subsequent to a Design Basis Accident.

Specifications

The following specifications apply except during low-temperature physics tests.

A. SAFETY INJECTION AND RESIDUAL HEAT REMOVAL SYSTEMS

1. The reactor shall not be made critical except for low-temperature physics tests, unless the following conditions are met:

   a. The refueling water storage tank contains not less than 345,000 gallons of water with a boron concentration of at least 2000 ppm.

   b. Deleted

   c. The four accumulators are pressurized to a minimum of 598 psig and a maximum of 685 psig and each contains a minimum of 723 ft^3 and a maximum of 875 ft^3 of water with a boron concentration of at least 2000 ppm. None of these four accumulators may be isolated.

   d. Three safety injection pumps together with their associated piping and valves are operable.

   e. Two residual heat removal pumps and heat exchangers together with their associated piping and valves are operable.

f. Two recirculation pumps together with the associated piping and valves are operable.

g. Valves 842 and 843 in the mini-flow return line from the discharge of the safety injection pumps to the RWST are de-energized in the open position.

h. Valves 856A, C, D and E, in the discharge header of the safety injection header, are in the open position. Valves 856B and F, in the discharge header of the safety injection header, are in the closed position. The hot-leg valves (856B and F) shall be closed with their motor operators de-energized by locking out the circuit breakers at the Motor Control Centers.

i. The four accumulator isolation valves shall be open with their motor operators de-energized by locking out the circuit breakers at the Motor Control Centers.

j. Valve 1810 on the suction line of the high-head SI pumps and valves 882 and 744, respectively on the suction and discharge line of the residual heat removal pumps, shall be blocked open by de-energizing the valve-motor operators.

k. The refueling water storage tank low-level alarms are operable and set to alarm between 74,200 gallons and 99,000 gallons of water in the tank.

2. During power operation, the requirements of 3.3.A.1 may be modified to allow any one of the following components to be inoperable at any one time. If the system is not restored to meet the requirements of 3.3.A.1 within the time period specified, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If the requirements of 3.3.A.1 are not satisfied within an additional 48 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

a. One safety injection pump may be out of service, provided the pump is restored to operable status within 24 hours and the remaining two pumps are operable.

b. One residual heat removal pump may be out of service, provided the pump is restored to operable status within 24 hours and the other residual heat removal pump is operable.

c. One residual heat removal heat exchanger may be out of service provided that it is restored to operable status within 48 hours.

d. Any valve required for the functioning of the system during and following accident conditions may be inoperable provided that it is restored to operable status within 24 hours and all valves in the system that provide the duplicate function are operable.

e. Deleted

f. One refueling water storage tank low-level alarm may be inoperable for up to 7 days provided the other low-level alarm is operable.

3. When RCS temperature is less than or equal to 305°F, the requirements of Table 3.1.A-2 regarding the number of safety injection (SI) pumps allowed to be energized shall be adhered to.

B. CONTAINMENT COOLING AND IODINE REMOVAL SYSTEMS

1. The reactor shall not be made critical unless the following conditions are met:

a. The spray additive tank contains not less than 4000 gallons of solution with a sodium hydroxide concentration of not less than 33% by weight.

b. The five fan cooler-charcoal filter units and the two spray pumps, with their associated valves and piping, are operable.

2. During power operation, the requirements of 3.3.B.1 may be modified to allow any one of the following components to be inoperable. If the system is not restored to meet the requirements of 3.3.B.1 within the time period specified, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If the requirements of 3.3.B.1 are not satisfied within an additional 48 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

   a. One fan cooler unit may be inoperable during normal reactor operation for a period not to exceed 7 days provided both containment spray pumps are operable.

   b. One containment spray pump may be inoperable during normal reactor operation, for a period not to exceed 72 hours, provided the five fan cooler units and the remaining containment spray pump are operable.

   c. Any valve required for the functioning of the system during and following accident conditions may be inoperable provided it is restored to operable status within 7 days or 72 hours for the fan cooler or containment spray systems respectively, and all valves in the system that provide the duplicate function are operable.

   d. The spray additive tank and its associated piping, valves and eductors may be inoperable during normal reactor operation for a period not to exceed 72 hours provided both containment spray pumps and the five fan cooler units are operable.

C. <u>ISOLATION VALVE SEAL WATER SYSTEM (IVSWS)</u>

1. The reactor shall not be brought above cold shutdown unless the following requirements are met:

   a. The IVSWS shall be operable.

   b. The IVSW tank shall be maintained at a minimum pressure of 52 psig and contain a minimum of 144 gallons of water.

2. The requirements of 3.3.C.1 may be modified to allow any one of the following components to be inoperable at any one time:

   a. Any one header of the IVSWS may be inoperable for a period not to exceed seven consecutive days.

   b. Any valve required for the functioning of the system during and following accident conditions may be inoperable provided it is restored to an operable status within seven days and all valves in the system that provide a duplicate function are operable.

3. If the IVSWS System is not restored to an operable status within the time period specified, then:

   a. If the reactor is critical, it shall be brought to the hot shutdown condition utilizing normal operating procedures. The shutdown shall start not later than at the end of the specified time period.

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

   c. In either case, if the IVSW System is not restored to an operable status within an additional 48 hours, the reactor shall be brought to the cold shutdown condition utilizing normal operating procedures. The shutdown shall start no later than the end of the 48-hour period.

D. <u>WELD CHANNEL AND PENETRATION PRESSURIZATION SYSTEM (WC & PPS)</u>

1. The reactor shall not be brought above cold shutdown unless:

   a. All required portions of the four WC & PPS zones are pressurized at or above 47 psig.

   b. The uncorrected air consumption for the WC & PPS is less than or equal to 0.2% of the containment volume per day.

2. The requirements of 3.3.D.1 may be modified as follows:

a. Any one zone of the WC & PPS may be inoperable for a period not to exceed seven consecutive days.

b. The uncorrected air consumption for the WC & PPS may be in excess of 0.2% of the containment volume per day for a period not to exceed seven consecutive days.

c. With the portion of the weld channel pressurization system inoperable, and it is determined that it is not repairable by any practicable means, then that portion may be disconnected from the system.

3. If the WC & PP System is not restored to an operable status within the time period specified, then:

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

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

c. In either case, if the WC & PP System is not restored to an operable status within an additional 48 hours, the reactor shall be brought to the cold shutdown condition utilizing normal operating procedures. The shutdown shall start no later than the end of the 48-hour period.

E. <u>COMPONENT COOLING SYSTEM</u>

1. The reactor shall not be made critical unless the following conditions are met:

a. Three component cooling pumps together with their associated piping and valves are operable.

b. Two auxiliary component cooling pumps together with their associated piping and valves are operable.

c. Two component cooling heat exchangers together with their associated piping and valves are operable.

2. During power operation, the requirements of 3.3.E.1 may be modified to allow one of the following components to be inoperable at any one time. If the system is not restored to meet the conditions of 3.3.E.1 within the time period specified, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If the requirements of 3.3.E.1 are not satisfied within an additional 48 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

a. One of the three operable component cooling pumps may be out of service provided the pump is restored to operable status within 14 days.

b. An additional component cooling pump may be out of service provided a second pump is restored to operable status within 24 hours.

c. One auxiliary component cooling pump may be out of service provided the pump is restored to operable status within 24 hours and the other pump is operable.

d. One component cooling heat exchanger or other passive component may be out of service for a period not to exceed 48 hours provided the system may still operate at design accident capability.

F. <u>SERVICE WATER SYSTEM</u>

1. DESIGNATED ESSENTIAL HEADER

a. The reactor shall not be above 350°F unless three service water pumps with their associated piping and valves are operable on the designated essential header.

b. When the reactor is above 350°F and one of the three service water pumps or any of its associated piping or valves is found inoperable,

and an essential service water header that meets the requirements of 3.3.F.1.a. cannot be restored within 12 hours, the reactor shall be placed in the hot shutdown condition within the next 6 hours and subsequently cooled below 350°F using normal operating procedures.

2. DESIGNATED NON-ESSENTIAL HEADER

a. The reactor shall not be above 350°F unless two service water pumps with their associated piping and valves are operable on the designated non-essential header.

b. When the reactor is above 350°F and one of the two service water pumps or any of its associated piping or valves is found inoperable, and a non-essential service water header that meets the requirements of 3.3.F.2.a cannot be restored within 24 hours, the reactor shall be placed in the hot shutdown condition within the next 6 hours and subsequently cooled below 350°F using normal operating procedures.

3. INTERCONNECTION OF HEADERS

Isolation shall be maintained between the essential and non-essential headers at all times when the reactor is above 350°F except for a period of up to 8 hours when the header may be connected to facilitate safety-related activities.

4. SERVICE WATER INLET TEMPERATURE

a. The reactor shall not be above 350°F unless the service water inlet temperature is less than or equal to 95°F, or

b. When the reactor is above 350°F and the service water inlet temperature exceeds 95°F, the reactor shall be placed in the hot shutdown condition within the next 7 hours and subsequently cooled below 350°F using normal operating procedures.

c. The provisions of Specification 3.0.1 do not apply.

5. SERVICE WATER INLET TEMPERATURE MONITORING INSTRUMENTATION

a. The service water inlet temperature monitoring instrumentation shall measure the Hudson River water temperature at the Indian Point Unit No. 2 intake structure,

b. The service water inlet temperature monitoring instrumentation shall be operable when intake water temperature, averaged over a 24 hour period, reaches 80°F, and when the reactor is above 350°F,

c. When the requirements of Specification 3.3.F.5.b apply, temperature measurements shall be taken every 4 hours up to and including a service water inlet temperature of 90°F; when the service water inlet temperature exceeds 90°F, temperature measurements shall be taken once an hour,

d. If the service water inlet temperature monitoring instrumentation is declared inoperable, it shall be either restored to operable status or alternative measurements shall be taken with a calibrated portable instrument within the applicable measurement time frame requirements of Specification 3.3.F.5.c, and

e. If the requirements of Specification 3.3.F.5.d cannot be met, the reactor shall be placed in the hot shutdown condition within the next 7 hours and subsequently cooled below 350°F using normal operating procedures.

G. <u>HYDROGEN RECOMBINER SYSTEM AND POST-ACCIDENT CONTAINMENT VENTING SYSTEM</u>

1. The reactor shall not be made critical unless the following conditions are met:

a. Both hydrogen recombiner units together with their associated piping, valves, oxygen supply system and control system are operable, with the exception of one recombiner unit's equipment located outside the containment which may be inoperable, provided it is under repair and can be made operable if needed.

b. The post-accident containment venting system is operable.

c. Hydrogen and oxygen supplies shall not be connected to the hydrogen recombiner units except under conditions of an accident or those specified in Specification 4.5.C.1.

2. During power operation, the requirements of 3.3.G.1 may be modified to allow any one of the following components to be inoperable. If the system is not restored to meet the requirements of 3.3.G.1 within the time specified, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures.

a. One hydrogen recombiner unit or its associated flow path, or oxygen supply system or control system may be inoperable for a period not to exceed thirty days, provided the other recombiner unit and the post-accident containment venting system are operable.

b. The post-accident containment venting system may be inoperable for a period not to exceed thirty days provided that both hydrogen recombiners are operable.

H. CONTROL ROOM AIR FILTRATION SYSTEM

1. The control room air filtration system shall be operable at all times when containment integrity is required.

2. From the date that the control room air filtration system becomes and remains inoperable for any reason, operations requiring containment integrity are permissible only during the succeeding 3.5 days. At the end of this 3.5 days period, if the conditions for the control room air filtration system cannot be met, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If the conditions are not satisfied within an additional 48 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

3. Two independent toxic gas detection systems, each capable of detecting chlorine and anhydrous ammonia shall be operable at all times except as

specified in 3.a, 3.b, or 3.c below. The alarm/trip setpoints for the chlorine and anhydrous ammonia gas detection systems shall be adjusted to actuate at a toxic gas concentration of less than or equal to 3.5 ppm and 25 ppm, respectively.

a. With one toxic gas detection system inoperable, restore the inoperable detection system to operable status within 7 days.

b. If 3.a above cannot be satisfied within the specified time, then, within the next 6 hours, initiate and maintain operation of the control room ventilation system in the recirculation mode of operation.

c. With both toxic gas detection systems inoperable for any one toxic gas, within one hour initiate and maintain operation of the control room ventilation in the recirculation mode of operation.

I. CABLE TUNNEL VENTILATION FANS

1. The reactor shall not be made critical unless the two cable tunnel ventilation fans are operable.

2. During power operation, the requirement of 3.3.I.1 may be modified to allow one cable tunnel ventilation fan to be inoperable for seven days, provided the other fan is operable.

Basis

The normal procedure for starting the reactor is, first, to heat the reactor coolant to near operating temperature by running the reactor coolant pumps. The reactor is then made critical by withdrawing control rods and/or diluting boron in the coolant[1]. With this mode of start-up, the energy stored in the reactor coolant during the approach to criticality is substantially equal to that during power operation, and therefore the minimum required engineered safeguards and auxiliary cooling systems are required to be operable. During low-temperature physics tests there is a negligible amount of stored energy in the reactor coolant; therefore, an accident comparable in severity to the Design Basis Accident is not possible, and the engineered safeguards systems are not required.

When the reactor is critical, the probability of sustaining both a major accident and a simultaneous failure of a safeguards component to operate as designed is necessarily very small. Thus operation with the reactor critical with minimum safeguards operable for a limited period does not significantly increase the probability of an accident having consequences which are more severe than the Design Basis Accident.

The operable status of the various systems and components is to be demonstrated by periodic tests, defined by Specification 4.5. A large fraction of these tests will be performed while the reactor is operating in the power range. If a component is found to be inoperable, it will be possible in most cases to effect repairs and restore the system to full operability within a relatively short time. Inoperability of a single component does not negate the ability of the system to perform its function[2], but it reduces the redundancy provided in the reactor design and thereby limits the ability to tolerate additional equipment failures. To provide maximum assurance that the redundant component(s) will operate if required to do so, the redundant component(s) are to be tested prior to initiating repair of the inoperable component. If it develops that (1) the inoperable component is not repaired within the specified allowable time period, or (2) a second component in the same or related system is found to be inoperable, the reactor will initially be put in the hot shutdown condition to provide for reduction of the decay heat from the fuel and consequent reduction of cooling requirements after a postulated loss-of-coolant accident. This will also permit improved access for repairs in some cases. After a limited time in hot shutdown, if the malfunction(s) are not corrected, the reactor will be placed in the cold shutdown condition, utilizing normal shutdown and cooldown procedures. In the cold shutdown condition there is no possibility of an accident that would release fission products or damage the fuel elements.

The plant operating procedures require immediate action to effect repairs of an inoperable component, and therefore in most cases repairs will be completed in less than the specified allowable repair times. The specified repair times do not apply to regularly scheduled maintenance of the engineered safeguards systems, which is normally to be performed during refueling shutdowns. The limiting times to repair are based on two considerations:

1. assurance with high reliability that the safeguard system will function properly if required to do so, and

2. allowance of sufficient time to effect repairs using safe and proper procedures.

Assuming the reactor has been operating at full-rated power for at least 100 days, the magnitude of the decay heat decreases after initiating hot shutdown. Thus the requirement for core cooling in case of a postulated loss-of-coolant accident while in the hot shutdown condition is significantly reduced below the requirements for a postulated loss-of-coolant accident during power operation. Putting the reactor in the hot shutdown condition significantly reduces the potential consequences of a loss-of-coolant accident, and also allows more free access to some of the engineered safeguards components in order to effect repairs.

Failure to complete repairs within 48 hours of going to the hot shutdown condition is considered indicative of a requirement for major maintenance, and therefore in such a case the reactor is to be put into the cold shutdown condition.

Valves 1810, 744 and 882 are kept in the open position during plant operation to assure that flow passage from the refueling water storage tank will be available during the injection phase of a loss-of-coolant accident. As an additional assurance of flow passage availability, the valve motor operators are de-energized to prevent an extremely unlikely spurious closure of these valves to take place. This additional precaution is acceptable since failure to manually re-establish power to close valves 1810 and 882, following the injection phase, is tolerable as a single failure. Valve 744 will not need to be closed following the injection phase. The accumulator isolation valve motor operators are de-energized to prevent an extremely unlikely spurious closure of these valves from occurring when accumulator core cooling flow is required.

With respect to the core cooling function, there is some functional redundancy for certain ranges of break sizes. The measure of effectiveness of the Safety Injection System is the ability of the pumps and accumulators to keep the core flooded or to reflood the core rapidly where the core has been uncovered for postulated large area ruptures. The result of the performance is to sufficiently limit any increase in clad temperature below a value where emergency core cooling objectives are met[9]. The range of core protection as a function of break diameter provided by the various components of the Safety Injection System is presented in Figure 6.2-9 of the UFSAR.

The requirement regarding the maximum number of SI pumps that can be energized when RCS temperature is less than or equal to 305°F is discussed under Specification 3.1.A.

The containment cooling and iodine removal functions are provided by two independent systems: (1) fan-coolers plus charcoal filters and (2) containment spray with sodium hydroxide addition. During normal power operation, the five fan-coolers are required to remove heat lost from equipment and piping within containment at design conditions (with a cooling water temperature of 95°F)[12].
In the event of a Design Basis Accident, any one of the following combinations will provide sufficient cooling to reduce containment pressure at a rate consistent with limiting offsite doses to acceptable values: (1) five fan-cooler units, (2) two containment spray pumps, (3) three fan-cooler units and one spray pump. Also, in the event of a Design Basis Accident, three charcoal filters (and their associated recirculation fans) in operation, along with one containment spray pump and sodium hydroxide addition, will reduce airborne organic and molecular iodine activities sufficiently to limit offsite doses to acceptable values. These constitute the minimum safeguards for iodine removal, and are capable of being operated on emergency power with one diesel generator inoperable.

If offsite power is available or all diesel generators are operating to provide emergency power, the remaining installed iodine removal equipment (two charcoal filters and their associated fans, and one containment spray pump and sodium hydroxide addition) can be operated to provide iodine removal in excess of the minimum requirements. Adequate power for operation of the redundant containment heat removal systems (i.e., five fan-cooler units or two containment spray pumps) is assured by the availability of offsite power or operation of all emergency diesel generators.

One of the five fan cooler units is permitted to be inoperable during power operation. This is an abnormal operating situation, in that the normal plant operating procedures require that an inoperable fan-cooler be repaired as soon as practical.

However, because of the difficulty of gaining access to make repairs, it is important on occasion to be able to operate temporarily without at least one fan-cooler. Compensation for this mode of operation is provided by the high degree of redundancy of containment cooling systems during a Design Basis Accident.

The Component Cooling System is different from the system discussed above in that the pumps are so located in the Auxiliary Building as to be accessible for repair after a loss-of-coolant accident[6]. During the recirculation phase following a loss-of-coolant accident, only one of the three component cooling pumps is required for minimum safeguards[7]. With two operable component cooling pumps, 100% redundancy will be provide. A total of three operable component cooling pumps will provide 200%

redundancy. The 14 day out of service period for the third component cooling pump is allowed since this is the 200% redundant pump.

A total of six service water pumps are installed. Only two of the set of three service water pumps on the header designated the essential header are required immediately following a postulated loss-of-coolant accident[8]. The limit on the service water maximum inlet temperature assures that the service water and component cooling water systems will be able to dissipate the heat loads generated in the limiting design basis accident.[12]

During the second phase of the accident, one additional service water pump on the non-essential header will be manually started to supply the minimum cooling water requirements for the component cooling loop.

The limits for the accumulators and their pressure and volume assure the required amount of water injection following a loss-of-coolant accident, and are based on the values used for the accident analysis[9].

Two independent diverse systems are provided for removal of combustible hydrogen from the containment building atmosphere: (1) the hydrogen recombiners, and (2) the post-accident containment venting system. Either of the two (2) hydrogen recombiners or the post-accident containment venting system are capable of wholly providing this function in the event of a design basis accident.

Two full-rated hydrogen recombination systems are provided in order to control the hydrogen evolved in the containment following a loss-of-coolant accident. Either system is capable of preventing the hydrogen concentration from exceeding 2% by volume within the containment. Each system is separate from the other and is provided with redundant features. Power supplies for the blowers and ignitors are separate, so that loss of one power supply will not affect the remaining system. Hydrogen gas is used as the externally supplied fuel. Oxygen gas is added to the containment atmosphere through a separate containment feed to prevent depletion of oxygen in the air below the concentration required for stable operation of the combustor (12%). The containment atmosphere sampling system consists of a sample line which originates in each of the containment fan cooler units. The fan and sampling pump head together are sufficient to pump containment air in a loop from the fan cooler through a containment penetration to a sample vessel outside the containment, and then through a second penetration to the sample termination inside the containment. The design hydrogen concentration for operating the recombiner is established at 2% by volume.

Conservative calculations indicate that the hydrogen content within the containment will not reach 2% by volume until 13 days after a loss-of-coolant accident. There is therefore no need for immediate operation of the recombiner following an accident, and the quantity of hydrogen fuel stored at the site will be only for periodic testing of the recombiners.

The Post-Accident Containment Venting System consists of a common penetration line which acts as a supply line through which hydrogen-free air can be admitted to the containment, and an exhaust line, with parallel valving and piping, through which hydrogen-bearing gases from containment may be vented through a filtration system.

The supply flow path makes use of instrument air to feed containment. The nominal flow rate from either of the two instrument air compressors is 200 scfm. If the instrument air system is not available, the station air system is available as a backup.

The exhaust line penetrates the containment and then is divided into two parallel lines. Each parallel line contains a pressure sensor and all the valves necessary for controlling the venting operation. The two lines then rejoin and the exhaust passes through a flow sensor and a temperature sensor before passing through roughing, HEPA and charcoal filters. The exhaust is then directed to the plant vent.

The post-accident containment venting system is a passive system in the sense that a differential pressure between the containment and the outside atmosphere provides the driving force for the venting process to take place. The system is designed such that a minimum internal containment pressure of 2.14 psig is required for the system to operate properly.

The flow rate and the duration of venting required to maintain the hydrogen concentration at or below 3 percent of the containment volume are determined from the containment hydrogen concentration measurements and the hydrogen generation rate. The containment pressure necessary to obtain the required vent flow is then determined. Using one of the air compressors, hydrogen-free air is pumped into the containment until the required containment pressure is reached. The air supply is then stopped and the supply/exhaust line is isolated by valves outside the containment. The addition of air to pressurize the containment dilutes the hydrogen; therefore, the containment will remain isolated until analysis of samples indicates that the concentration is again approaching 3 percent by volume. Venting will then be started. This process of containment pressurization followed by venting is repeated as may be necessary to maintain the hydrogen concentration at or below 3 volume percent.

The post-accident venting system is used only in the absence of hydrogen recombiners and only when absolutely necessary. From the standpoint of minimizing offsite radiation doses, the optimum starting time for the venting system, if needed, is the latest possible time after the accident. Consistent with this philosophy, the selected venting initiation point of 3 percent hydrogen maximizes the time period before venting is required while at the same time allows a sufficient margin of safety below the lower flammability limit of hydrogen.

The control room air filtration system is designed to filter the control room atmosphere for intake air and/or for recirculation during control room isolation conditions. The control room system is designed to automatically start upon control room isolation. Control room isolation is initiated either by a safety injection signal or by detection of high radioactivity in the control room. If the control room air filtration system is found to be inoperable, there is no immediate threat to the control room and reactor operation may continue for a limited period of time while repairs are being made. If the system cannot be repaired within 3.5 days, the reactor is placed in the hot shutdown condition.

The control room ventilation system is equipped with toxic gas detection systems consisting of redundant monitors capable of detecting chlorine and anhydrous ammonia. These toxic gas detection systems are designed to isolate the control room from outside air upon detection of toxic concentration of the monitored gases in the control room ventilation system. The operability of the toxic gas detection systems provides assurance that the control room operators will have adequate time to take protective action in the event of an accidental toxic gas release. Selection of the gases to be monitored and the setpoint established for the monitors are based on the results described in the Indian Point Unit No. 2 Control Room Habitability Study dated June 10, 1991.

The cable tunnel is equipped with two temperature-controlled ventilation fans. Each fan has a capacity of 21,000 cfm and is connected to a 480v bus. One fan will start automatically when the temperature in the tunnel reaches 100°F. Under the worst conditions, i.e., loss of outside power and all the Engineered Safety Features in operation, one ventilation fan is capable of maintaining the tunnel temperature below 104°F. Under the same worst conditions, if no ventilation fans were operating, the natural air circulation through the tunnel would be sufficient to limit the gross tunnel temperature to below the tolerable value of 140°F. However, in order to provide for

ample tunnel ventilation capacity, the two ventilation fans are required to be operable when the reactor is made critical. If one ventilation fan is found inoperable, the other fan will ensure that cable tunnel ventilation is available.

Valves 856A, C, D and E are maintained in the open position during plant operation to assure a flow path for high-head safety injection during the injection phase of a loss-of-coolant accident. Valves 856B and F are maintained in the closed position during plant operation to prevent hot-leg injection during the injection phase of a loss-of-coolant accident. As an additional assurance of preventing hot-leg injection, the valve motor operators are de-energized to prevent spurious opening of these valves. Power will be restored to these valves at an appropriate time in accordance with plant operating procedures after a loss-of-coolant accident in order to establish hot-leg recirculation.

Valves 842 and 843 in the mini-flow return line from the discharge of the safety injection pumps to the refueling water storage tank are de-energized in the open position to prevent an extremely unlikely spurious closure which would cause the safety injection pumps to overheat if the reactor coolant system pressure is above the shutoff head of the pumps.

The specified quantities of water for the RWST include unavailable water (4687 gals) in the tank bottom, inaccuracies (24,800 gals) in the alarm setpoints, the minimum quantity required during the injection (246,000 gals)[12] for accident mitigation and the minimum quantity required during the recirculation phase (60,000 gals) for post-LOCA NaOH requirements inside containment. The minimum RWST inventory (i.e., 345,000 gals) provides approximately 9,500 gallons margin.

The seven-day out-of-service period for the Weld Channel and Penetration Pressurization System and the Isolation Valve Seal Water System is allowed because no credit has been taken for operation of these systems in the calculation of offsite accident doses should an accident occur. No other safeguards systems are dependent on operation of these systems[11]. The minimum pressure settings for the IVSWS and WC & PPS during operation assures effective performance of these systems and assures that the containment design pressure of 47 psig is not exceeded. Portions of the Weld Channel Pressurization System are in areas that are not accessible, such as below the concrete floor of containment or in high radiation areas. If it is determined that it is not practicable to repair an inoperable portion of the system, then that portion may be disconnected.

References

(1) UFSAR Section 9

(2) UFSAR Section 6.2

(3) DELETED

(4) UFSAR Section 6.4

(5) Reference Deleted

(6) UFSAR Section 9.3

(7) UFSAR Section 9.3

(8) UFSAR Section 9.6.1

(9) UFSAR Section 14.3

(10) Indian Point Unit No. 2, UFSAR Sections 6.2 and 6.3 and the Safety Evaluation accompanying "Application for Amendment to Operating License" sworn to by Mr. William J. Cahill, Jr. on March 28, 1977.

(11) UFSAR Sections 6.5 and 6.6

(12) WCAP-12312, "Safety Evaluation for An Ultimate Heat Sink Temperature to 95°F at Indian Point Unit 2", July, 1989.

3.6 CONTAINMENT SYSTEM

Applicability

Applies to the integrity of reactor containment.

Objective

To define the operating status of the reactor containment for plant operation.

Specifications

A. CONTAINMENT INTEGRITY

1. The following requirements shall be satisfied: (a) whenever the reactor is above cold shutdown or (b) whenever the reactor vessel head is less than fully tensioned, and (i) the shutdown margin is <5% $\Delta k/k$, or (ii) the boron concentration within the reactor is less than 2000 ppm.

   a. All non-automatic containment isolation valves which are not required to be open during accident conditions are closed and blind flanges installed where required. Those non-automatic containment isolation valves listed in Table 3.6-1 and any test connection valves which are located between containment isolation valves and which are normally closed with threaded caps or blind flanges installed, may be opened if necessary for plant operation or for testing and only as long as necessary to perform the intended function.

   b. All automatic containment isolation valves are either operable or in the closed position or isolated by a closed manual valve or flange that meets the same design criteria as the isolation valve.

   c. The equipment door is properly closed.

   d. At least one door in each personnel air lock is properly closed.

   e. The WC&PPS requirements of Specification 3.3.D are being satisfied.

f. Containment leakage has been verified in accordance with the surveillance requirements of Specification 4.4.

2. The following additional requirements shall be satisfied during power operation:

   a. The automatic containment purge and containment pressure relief isolation valves are set to limit valve disk travel to no greater than 60° open (90° being full open) with stroke times of three seconds or less.

   b. The automatic containment purge and containment pressure relief isolation valves may only be open for safety-related reasons.[1)]

3. Except as specified in 3.a. below, if the above requirements are not satisfied, the condition shall be corrected within 4 hours or the reactor shall be brought to a cold shutdown condition within the next 36 hours, utilizing normal operating procedures.

   a. With one or more isolation valve(s) inoperable:

      1. maintain at least one isolation valve operable in each affected penetration[2)], and

      2. either:

         (a) Restore the inoperable valve(s) to operable status within 4 hours, or

         (b) Isolate each affected penetration within 4 hours by use of at least one deactivated automatic isolation valve secured in the isolation position[3)], or

---

1) Examples of safety-related reasons include containment pressure control, or to facilitate safety-related surveillance or safety-related maintenance.

2) Not required for penetrations equipped with only one isolation valve.

3) This may be the valve previously maintained operable per 3.a.1 above or the valve initially declared inoperable.

(c) Isolate each affected penetration within 4 hours by use of at least one closed manual valve[3] or blind flange that meets the design criteria for an isolation valve, or

(d) Be in cold shutdown within the following 36 hours, utilizing normal operating procedures.

4. Non-automatic containment isolation valves may be added to plant systems, without prior license amendment to Table 3.6-1, provided that a revision to this Table is included in a subsequent license amendment application.

B. INTERNAL PRESSURE

If the internal pressure exceeds 2 psig or the internal vacuum exceeds 2.0 psig, the condition shall be corrected or the reactor shut down.

C. CONTAINMENT TEMPERATURE

The reactor shall not be taken above the cold shutdown condition unless the containment ambient temperature is greater than 50°F.

Basis

The Reactor Coolant System conditions of cold shutdown assure that no steam will be formed and hence there would be no pressure buildup in the containment if a Reactor Coolant System rupture were to occur.

The shutdown margins are selected based on the type of activities that are being carried out. The shutdown margin requirement of Specification 3.8.B.2 when the head is off precludes criticality during refueling. When the reactor head is not to be removed, the specified cold shutdown margin of 1% $\Delta k/k$ precludes criticality at cold shutdown conditions.

Regarding internal pressure limitations, the containment calculated peak accident pressure of 47 psig would not be exceeded if the internal pressure before a major loss-of-coolant accident were as much as 8 psig. The containment can withstand an internal vacuum of 2.5 psig. The 2.0 psig vacuum specified as an operating limit avoids any difficulties with motor cooling.

The requirement of a 50°F minimum containment ambient temperature is to assure that the minimum service metal temperature of the containment liner is well above the NDT + 30°F criterion for the liner material[1].

Table 3.6-1 lists non-automatic valves that are designated as part of the containment isolation function. During periods of normal plant operations requiring containment integrity, valves on this Table will be open either continuously or intermittently depending on requirements of the particular protection, safeguards or essential service systems. The valves to be open intermittently are under administrative control and are open only for as long as necessary to perform their intended function. In all cases, however, the valves listed in Table 3.6-1 are closed during the post-accident period in accordance with plant procedures and consistent with requirements of the related protection, safeguards, or essential service systems.

Reference

(1) UFSAR Section 5.1.1.1

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Applicability

Applies to the limits on core fission power distributions and to the limits on control rod operations.

Objectives

1. to ensure core subcriticality after reactor trip,

2. to ensure acceptable core power distribution during power operation in order to maintain fuel integrity in normal operation and transients associated with faults of moderate frequency, supplemented by automatic protection and by administrative procedures, and to maintain the design basis initial conditions for limiting faults, and

3. to limit potential reactivity insertions caused by hypothetical control rod ejection.

Specifications

3.10.1 Shutdown Reactivity

The shutdown margin shall be at least as great as that shown in Figure 3.10-1.

3.10.2 Power Distribution Limits

3.10.2.1 At all times, except during low-power physics tests, the hot channel factors defined in the basis must meet the limits specified in the Core Operating Limits Report (COLR).

3.10.2.2 Following initial core loading, subsequent reloading and at regular effective full-power monthly intervals thereafter, power distribution maps, using the movable detector system, shall be made to confirm that the hot channel factor limits of the COLR are satisfied. For the purpose of this comparison,

3.10.2.2.1 The measurement of total peaking factor, F_Q^{Meas}, shall be increased by three percent to account for manufacturing tolerances and further increased by five percent to account for measurement error.

3.10.2.2.2 The measurement of enthalpy rise hot channel factor, $F^N{}_\Delta H$, shall be increased by four percent to account for measurement error. If either measured hot channel factor exceeds its limit specified in the COLR, the reactor power and high neutron flux trip setpoint shall be reduced so as not to exceed a fraction of rated value equal to the ratio of the F_Q or $F^N{}_\Delta H$ limit to measured value, whichever is less. If subsequent in-core mapping cannot, within a 24-hour period, demonstrate that the hot channel factors are met, the reactor shall be brought to a hot shutdown condition with return to power authorized only for the purpose of physics testing.

3.10.2.3 The reference equilibrium indicated axial flux difference as a function of power level (called the target flux difference) shall be measured at least once per effective full-power quarter. The target flux difference must be updated each effective full-power month by linear interpolation using the most recent measured value and a value of approximately 0 percent at the end of the cycle life.

3.10.2.4 Except during physics tests, during excore calibration procedures and except as modified by Items 3.10.2.5 through 3.10.2.7 below, the indicated axial flux difference shall be maintained within the band specified in the COLR about the target flux difference (defines the band on axial flux difference).

3.10.2.5 At a power level greater than 90% of rated power,

3.10.2.5.1 If the indicated axial flux difference deviates from its target band, the flux difference shall be returned to its target band immediately or the reactor power shall be reduced to a level no greater than 90 percent of rated power.

3.10.2.6 At a power level no greater than 90 percent of rated power,

3.10.2.6.1 The indicated axial flux difference may deviate from its target band specified in the COLR for a maximum of one hour (cumulative) in any 24-hour period provided the flux difference does not exceed an envelope bounded by that specified in the COLR at 90% power and increasing by

the value specified in the COLR for each 2 percent of rated power below 90% power.

3.10.2.6.2 If Specification 3.10.2.6.1 is violated, then the reactor power shall be reduced immediately to no greater than 50% power and the high neutron flux setpoint reduced to no greater than 55 percent of rated values.

3.10.2.6.3 A power increase to a level greater than 90 percent of rated power is contingent upon the indicated axial flux difference being within its target band.

3.10.2.7 At a power level no greater than 50 percent of rated power,

3.10.2.7.1 The indicated axial flux difference may deviate from its target band.

3.10.2.7.2 A power increase to a level greater than 50 percent of rated power is contingent upon the indicated axial flux difference not being outside its target band for more than two hours (cumulative) out of the preceding 24-hour period. One-half the time the indicated axial flux difference is out of its target band up to 50% of rated power is to be counted as contributing to the one hour cumulative maximum the flux difference may deviate from its target band at a power level ≤ 90% of rated power.

3.10.2.8 Alarms are provided to indicate non-conformance with the flux difference requirements of 3.10.2.5.1 and the flux difference-time requirements of 3.10.2.6.1. If the alarms are temporarily out of service, conformance with the applicable limit shall be demonstrated by logging the flux difference at hourly intervals for the first 24 hours and half-hourly thereafter.

3.10.2.9 If the core is operating above 75% power with one excore nuclear channel out of service, then core quadrant power balance shall be determined once a day using movable incore detectors (at least two thimbles per quadrant).

3.10.3 <u>Quadrant Power Tilt Limits</u>

3.10.3.1 Except for physics tests, when the core is operating above 50% of rated thermal power and the indicated quadrant power tilt ratio exceeds 1.02 but is less than or equal to 1.09, within two hours reduce the quadrant power tilt ratio to within its limit or the following actions shall be taken:

a. Restrict core power level and reset the power range high flux setpoint three percent of rated values for every percent of indicated power tilt ratio exceeding 1.0, and

b. Verify that the quadrant power tilt ratio is within its limit within 24 hours after exceeding the limit or restrict core power level to less than 50% of rated thermal power within the next 2 hours and reduce the power range high flux trip setpoint to less than or equal to 55% of rated thermal power within the next 4 hours.

3.10.3.2 Except for physics tests, if the indicated quadrant power tilt ratio exceeds 1.09 with the core operating above 50% of rated thermal power and

a) there is a simultaneous indication of a misaligned control rod, restrict core power level three percent of rated value for every percent of indicated power tilt ratio exceeding 1.0 or until core power level is less than 50% of rated thermal power. If the quadrant power tilt ratio is not within its limit within 2 hours after exceeding the limit, restrict core power level to less than 50% of rated thermal power within the next 2 hours and reduce the power range high flux trip setpoint to less than or equal to 55% of rated thermal power within the next 4 hours.

-or-

b) there is no simultaneous indication of a misaligned control rod, reduce thermal power to less than 50% of rated thermal power within 2 hours and reduce the power range high flux trip setpoint to less than or equal to 55% of rated thermal power within the next 4 hours.

3.10.3.3 The rod position indicators shall be monitored and logged once each shift to verify rod position within each bank assignment.

3.10.3.4 The tilt deviation alarm shall be set to annunciate whenever the excore tilt ratio exceeds 1.02, except as modified in Specification 3.10.10.

3.10.4 Rod Insertion Limits

3.10.4.1 The shutdown rods shall be withdrawn as specified in the COLR when the reactor is critical or approaching criticality (i.e., the reactor is no longer subcritical by an amount equal to or greater than the shutdown margin in Figure 3.10-1).

3.10.4.2 When the reactor is critical, the control banks shall be limited in physical insertion to the insertion limits specified in the COLR.

3.10.4.3 Control bank insertion shall be further restricted if:

a. The measured control rod worth of all rods, less the worth of the most reactive rod (worst case stuck rod), is less than the reactivity required to provide the design value of available shutdown,

b. A rod is inoperable (Specification 3.10.7).

3.10.4.4 Insertion limits do not apply during physics tests or during periodic exercise of individual rods. *In addition, insertion limits do not apply when performing calibration of individual rod position indicator channels at or below the rating specified in the Core Operating Limits Report (COLR) but not higher than a nominal 30% power not to exceed 35% power. However, the shutdown margin indicated in Figure 3.10-1 must be maintained except for the low-power physics test to measure control rod worth and shutdown margin. For this test the reactor may be critical with all but one control rod inserted.

* Only for Cycle 13.

3.10.5 Rod Misalignment Limitations

3.10.5.1.1 If a control rod is misaligned from its bank demand position by more than ±12 steps when indicated control rod position is less than or equal to 210 steps withdrawn, then realign the rod or determine the core peaking factors within 2 hours and apply Specification 3.10.2.

3.10.5.1.2 If a control rod is misaligned from its bank demand position by more than +17, -12 steps when indicated control rod position is greater than or equal to 211 steps withdrawn, then realign the rod or determine the core peaking factors within 2 hours and apply Specification 3.10.2.

3.10.7 Inoperable Rod Limitations

3.10.7.1 An inoperable rod is a rod which does not trip or which is declared inoperable under Specification 3.10.5, or which fails to meet the requirements of Specification 3.10.8.

3.10.7.2 Not more than one inoperable control rod shall be allowed any time the reactor is critical except during physics tests requiring intentional rod misalignment. Otherwise, the plant shall be brought to the hot shutdown condition.

3.10.7.3 If any rod has been declared inoperable, then the potential ejected rod worth and associated transient power distribution peaking factors shall be determined by analysis within 30 days. The analysis shall include due allowance for non-uniform fuel depletion in the neighborhood of the inoperable rod. If the analysis results in a more limiting hypothetical transient than the cases reported in the safety analysis, the plant power level shall be reduced to an analytically determined part power level which is consistent with the safety analysis.

3.10.8 Rod Drop Time

At operating temperature and full flow, the drop time of each control rod shall be no greater than 2.4 seconds from gripper release to dashpot entry.

3.10.9 Rod Position Monitor

If the rod position deviation monitor is inoperable, individual rod positions shall be logged once per shift and after a load change greater than 10 percent of rated power.

3.10.10 Quadrant Power Tilt Monitor

If one or both of the quadrant power tilt monitors is inoperable, individual upper and lower excore detector calibrated outputs shall be logged once per shift and after a load change greater than 10 percent of rated power.

Basis

Design criteria have been chosen for normal operations, for operational transients and for those events analyzed in UFSAR Section 14.1 which are consistent with the fuel integrity analyses. These relate to fission gas release, pellet temperature and cladding mechanical properties. Also the minimum DNBR in the core must be greater than the safety limits DNBRs in normal operation or in short-term transients.

In addition to the above conditions, the peak linear power density must not exceed the limiting kw/ft values which result from the large break loss-of-coolant accident analysis based on the ECCS acceptance criteria limit of 2200°F. This is required to meet the initial conditions assumed for a loss-of-coolant accident. To aid in specifying the limits on power distribution the following hot channel-factors are defined.

$F_Q(Z)$, Height Dependent Heat Flux Hot Channel Factor is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

F^EQ, Engineering Heat Flux Hot Channel Factor is defined as the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, surface area of the fuel rod and eccentricity of the gap between pellet and clad. Combined statistically the net effect is a factor of 1.03 to be applied to fuel rod surface heat flux.

$F^N\Delta H$, Nuclear Enthalpy Rise Hot Channel Factor is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

It should be noted that $F^N\Delta H$ is based on an integral and is used as such in the DNB calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizontal (x-y) power shapes throughout the core. Thus the horizontal power shape at the point of maximum heat flux is not necessarily directly related to $F^N\Delta H$.

The upper bound envelope of the total peaking factor (F_Q) specified in the COLR times the normalized peaking factor axial dependence of K(Z) specified in the COLR has been determined from extensive analyses considering all operating maneuvers consistent with the technical specifications on power distribution control as given in Section 3.10. The results of the loss-of-coolant accident analyses based on the specified F_Q times K(Z) specified in the COLR indicate a peak clad temperature of less than 2200°F for the worst case double-ended cold leg guillotine break.

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate allowance for a full core map taken with the movable incore detector flux mapping system and three percent is the appropriate allowance for manufacturing tolerance.

In the specified limit of $F^N_\Delta H$ there is an 8 percent allowance for uncertainties which means that normal operation of the core is expected to result in $F^N_\Delta H$ within the limits specified in the COLR. The logic behind the larger uncertainty in this case is that (a) normal perturbations in the radial power shape (e.g., rod misalignment) affect $F^N_\Delta H$, in most cases without necessarily affecting F_Q, (b) the operator has a direct influence on F_Q through movement of rods and can limit it to the desired value (he has no direct control over $F^N_\Delta H$) and (c) an error in the predictions for radial power shape, which may be detected during startup physics tests can be compensated for in F_Q by tighter axial control, but compensation for $F^N_\Delta H$ is less readily available. When a measurement of $F^N_\Delta H$ is taken, experimental error must be allowed for and 4 percent is the appropriate allowance for a full core map taken with the movable incore detector flux mapping system.

Measurements of the hot channel factors are required as part of startup physics tests at least each effective full-power month of operation, and whenever abnormal power distribution conditions require a reduction of core power to a level based on measured hot channel factors. The incore map taken following initial loading provides confirmation of the basic nuclear design bases, including proper fuel loading patterns. The periodic monthly incore mapping provides additional assurance that the nuclear design bases remain inviolate and identifies operational anomalies which would otherwise affect these bases.

For normal operation, it is not necessary to measure these quantities. Instead it has been determined that, provided certain conditions are observed, the hot channel factor limits will be met; these conditions are as follows:

1. Control rods in a single bank move together with no individual rod insertion differing by more than 15 inches from the bank demand position. An indicated misalignment limit of 12 steps precludes rod misalignment no greater than 15 inches with consideration of maximum instrumentation error for indicated rod position less than or equal to 210 steps withdrawn.

   For indicated control rod positions greater than or equal to 211 steps withdrawn, an indicated misalignment of +17 steps does not exceed the power peaking

factor limits. The reactivity worth of a rod at this core height (211 + steps) is not sufficient to perturb power shapes to the extent that peaking factors are affected.

2. At or below the rating specified in the Core Operating Limits Report (COLR) but no higher than 50% power the rod position indicator capability is less than or equal to 24 steps.

3. Control rod banks are sequenced with overlapping banks as described in Technical Specification 3.10.4.

4. The control rod bank insertion limits are not violated.

5. Axial power distribution control procedures, which are given in terms of flux difference control and control bank insertion limits, are observed. Flux difference refers to the difference in signals between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure of the axial offset which is defined as the difference in normalized power between the top and bottom halves of the core.

The permitted relaxation in $F^N\Delta H$ allows radial power shape changes with rod insertion to the insertion limits. It has been determined that, provided the above conditions (1 through 4) are observed, these hot channel factors limits are met. In the COLR, F_Q is arbitrarily limited for $P \leq 0.5$ (except for low-power physics tests).

The procedures for axial power distribution control referred to above are designed to minimize the effects of xenon redistribution on the axial power distribution during load-follow maneuvers. Basically, control of flux difference is required to limit the difference between the current value of Flux Difference (ΔI) and a reference value which corresponds to the full-power equilibrium value of Axial Offset (Axial Offset = ΔI/fractional power). The reference value of flux difference varies with power level and burnup, but, expressed as axial offset, it varies only with burnup.

The technical specifications on power distribution control assure that the total peaking factor upper-bound envelope of specified F_Q times K(Z) as specified in the COLR is not exceeded and xenon distributions are not developed which, at a later time, would cause greater local power peaking even though flux difference is then within the limits specified by the procedure.

The target (or reference) value of flux difference is determined as follows. At any time that equilibrium xenon conditions have been established, the indicated flux difference is noted with the control rod bank more than 190 steps withdrawn (i.e., normal full-power operating position appropriate for the time in life, usually withdrawn farther as burnup proceeds). This value, divided by the fraction of full-power at which the core was operating, is the full-power value of the target flux difference. Values for all other core power levels are obtained by multiplying the full-power value by the fractional power. Since the indicated equilibrium value was noted, no allowances for excore detector error are necessary and indicated axial flux difference deviation as specified in the COLR is permitted from the indicated reference value. During periods where extensive load following is required, it may be impractical to establish the required core conditions for measuring the target flux difference every month. For this reason, the specification provides two methods for updating the target flux difference.

Strict control of the flux difference (and rod position) is not as necessary during part-power operation. This is because xenon distribution control at part power is not as significant as the control at full power and allowance has been made in predicting the heat flux peaking factors for less strict control at part power. Strict control of the flux difference is not possible during certain physics tests or during required, periodic, excore calibrations which require larger flux differences than permitted. Therefore, the specifications on power distribution control are not applied during physics tests or excore calibrations; this is acceptable due to the low probability of a significant accident occurring during these durations.

In some instances of rapid plant power reduction, automatic rod motion will cause the flux difference to deviate from the target bank when the reduced power level is reached. This does not necessarily affect the xenon distribution sufficiently to change the envelope of peaking factors which can be reached on a subsequent return to full power within the target bank; however, to simplify the specification, a limitation of one hour in any period of 24 hours is placed on operation outside the band. This ensures that the resulting xenon distributions are not significantly different from those resulting from operation within the target band. The instantaneous consequence of being outside the band, provided rod insertion limits are observed, is not worse than a 10 percent increment in peaking factor for flux difference in the range specified in the COLR. Therefore, while the deviation exists, the power level is limited to 90 percent or less depending on the indicated flux difference.

If, for any reason, flux difference is not controlled within the limit specified in the COLR for as long a period as one hour, then xenon distributions may be significantly changed and operation at 50 percent is required to protect against potentially more severe consequences of some accidents.

As discussed above, the essence of the procedure is to maintain the xenon distribution in the core as close to the equilibrium full-power condition as possible. This is accomplished by using the boron system to position the control rods to produce the required indicated flux difference.

For Condition II events, the core is protected from overpower and a minimum DNBR of less than the safety limit DNBRs by an automatic protection system. Compliance with operating procedures is assumed as a precondition for Condition II transients; however, operator error and equipment malfunctions are separately assumed to lead to the cause of the transients considered.

Quadrant power tilt limits are based on the following considerations. Frequent power tilts are not anticipated during normal operation as this phenomenon is caused by some asymmetric perturbation, e.g., rod misalignment or inlet temperature mismatch. A dropped or misaligned rod will easily be detected by the Rod Position Indication system or core instrumentation per Specification 3.10.6, and core limits are protected per Specification 3.10.5. A quadrant tilt by some other means would not appear instantaneously but would build up over several hours, and the quadrant tilt limits are met to protect against this situation. They also serve as a backup protection against the dropped or misaligned rod. Operational experience shows that normal power tilts are less than 1.01. Thus, sufficient time is available to recognize the presence of a tilt and correct the cause before a severe tilt could build up. During startup and power escalation, however, a large tilt could be indicated. Therefore, the specification has been written so as to prevent escalation above 50 percent power if a large tilt is present. The numerical limits are set to be commensurate with design and safety limits for DNB protection and linear heat generation rate as described below.

The radial power distribution within the core must satisfy the design values assumed for calculation of power capability. Radial power distributions are measured as part of the startup physics testing and are periodically measured at a monthly or greater frequency. These measurements are taken to assure that the radial power distribution with any quarter core radial power asymmetry conditions are consistent with the assumptions used in power capability analyses. It is not intended that reactor operation would continue with a power tilt condition which exceeds the radial power asymmetry considered in the power capability analysis.

The quadrant tilt power deviation alarm is used to indicate a sudden or unexpected change from the radial power distribution mentioned above. The two percent tilt alarm setpoint represents a minimum practical value consistent with instrumentation errors and operating procedures. This asymmetry level is sufficient to detect significant misalignment of control rods. Misalignment of control rods is considered to be the most likely cause of radial power asymmetry. The requirement for verifying rod position once each shift is imposed to preclude rod misalignment which would cause a tilt condition of less than the 2% alarm level.

The two-hour time interval in this specification is considered ample to identify a dropped or misaligned rod and complete realignment procedures to eliminate the tilt condition. In the event that this tilt condition cannot be eliminated within the two-hour time allowance, additional time would be needed to investigate the cause of the tilt condition. The measurements would include a full-core physics map utilizing the movable detector system. For a tilt condition $\leq$ 1.09, an additional 22-hour time interval is authorized to accomplish these measurements. However, to assure that the peak core power is maintained below limiting values, a reduction of reactor power of three percent for each one percent of indicated tilt is required. Physics measurements have indicated that the core radial power peaking would not exceed a two to one relationship with the indicated tilt from the excore nuclear detector system for the worst rod misalignment.

In the event a tilt condition of $\leq$ 1.09 cannot be eliminated after 24 hours, the reactor power level will be reduced to less than 50% of rated power. To avoid reset of a large number of protection setpoints, the power range nuclear instrumentation would be reset to cause an automatic reactor trip at 55% of allowed power. A reactor trip at this power has been selected to prevent, with margin, exceeding core safety limits even with a nine percent tilt condition.

If a tilt ratio greater than 1.09 occurs, which is not due to a misaligned rod, the reactor power level will be reduced to less than 50% of rated power for investigation. However, if the tilt condition can be identified as due to rod misalignment, operation can continue at a reduced power (3% for each 1 percent the tilt ratio exceeds 1.0) for two hours to correct the rod misalignment.

Trip shutdown reactivity is provided consistent with plant safety analysis assumptions. One percent shutdown is adequate except for steam break analysis, which requires more shutdown if the boron concentration is low. Figure 3.10-1 is drawn accordingly.

Rod insertion limits are used to assure adequate trip reactivity, to assure meeting power distribution limits, and to limit the consequence of a hypothetical rod ejection accident. The available control rod reactivity, or excess beyond needs, decreases with decreasing boron concentration because the negative reactivity required to reduce the power level from full power to zero power is largest when the boron concentration is low.

Insertion limits do not apply during calibration of RPIs at or below the rating specified in the Core Operating Limits Report (COLR) but no higher than a nominal 30% power not to exceed 35% power because performing these calibrations at this reduced power ensures that the power peaking factor limits are met.

The intent of the test to measure control rod worth and shutdown margin (Specification 3.10.4) is to measure the worth of all rods less the worth of the worst case for an assumed stuck rod, that is, the most reactive rod. The measurement would be anticipated as part of the initial startup program and infrequently over the life of the plant, to be associated primarily with determinations of special interest such as end-of-life cooldown, or startup of fuel cycles which deviate from normal equilibrium conditions in terms of fuel loading patterns and anticipated control bank worths. These measurements will augment the normal fuel cycle design calculations and place the knowledge of shutdown capability on a firm experimental as well as analytical basis.

Operation with abnormal rod configuration during low-power and zero-power testing is permitted because of the brief period of the test and because special precautions are taken during these tests.

The primary means of determining the position of individual control rods is the Rod Position Indication system. The RPI system consists of an individual rod position detector mounted on the pressure housing of each of the rod drive mechanisms, rack mounted electronic equipment and indicating equipment mounted on the flight panel. The rod position detector is a linear variable transformer consisting of primary and secondary coils alternatively stacked on a stainless steel support tube. The mechanism drive shaft serves as a "core" of the transformer. With a constant AC source applied to the primary windings, the vertical position of the mechanism drive rod shaft changes the primary to secondary magnetic coupling and produces a unique AC secondary voltage. This output voltage is an analog signal which is proportional to the vertical position of the control rod. The AC output from the secondary coils is fed to the signal conditioning circuit on the rod position chassis where is it rectified to a DC signal and filtered. The resulting DC analog voltage which is proportional to rod position is fed to the following points.

a) Rod bottom bistable
b) Flight panel indicator
c) Position voltmeter on flight panel
d) Test points on front of chassis
e) Plant Computers

A zero and span adjustment is provided to produce an output voltage signal proportional to rod travel between rods full in and rods full out. Because there is only a zero and span adjustment, a two point calibration is done.

The rod position indicator channel is sufficiently accurate to detect a rod ±7.5 inches away from its demand position for indicated control rod position less than or equal to 210 steps withdrawn. An indicated misalignment ≤ 12 steps does not exceed the power peaking factor limits. A misaligned rod of + 17 steps allows for an instrumentation error of 12 steps plus 5 steps that are not indicated due to the location relationship of the RPI coil stack and the control rod drive rod for indicated rod position greater than or equal to 211 steps withdrawn. The last five steps of rod travel are not indicated by the RPI because the drive rod and spider assembly have been raised three inches (≈5 steps) from rod bottom. The reactivity worth of a rod at this core height (211 + steps) is not sufficient to perturb power shapes to the extent that peaking factors are affected.

Experience at Indian Point 2 and at other plants with similar RPI systems has shown that the output signal of the RPI is not exactly linear with respect to vertical position of the control rod. Thus, there is some inherent error initially in the RPI indication. However, by calibrating the shutdown bank and control banks A, B and C at the fully withdrawn position, and control bank D at its normal operating position, the calibration will be most accurate at the position where the rods are usually found. In addition, experience has shown that the proportionality constant is sensitive to temperatures.

As a result of the above an additional uncertainty is added to the normal measurement uncertainty. To account for these uncertainties, data points can be collected and an individual graph for each RPI can be provided to the operator. As an alternative to individual graphs, a larger total uncertainty can be assumed for the RPI along with an equivalent assumed misalignment of a rod from the bank demand position. Calculations have been done that demonstrate that a total of ±24 steps can be tolerated as an error at or below the reduced power level given in the COLR but no more than 50% power. Since at some power levels it is not possible to determine whether there is rod motion or the RPI has drifted or is inaccurate, the calculations have assumed in the worst case a misalignment of 48 steps between a D bank control rod and the remainder of its group (i.e., 24 steps due to the RPI indication and 24 steps

misalignment). This was also done for the C Bank (both banks were nominally at their 100% power insertion limits). For conservatism the Technical Specifications on allowed rod misalignment has been kept at ±12 steps, that is, for power levels where the rod position can be determined more accurately. If the indicated misalignment of ±24 steps has been exceeded, and a check has shown that the control rod(s) are indeed misaligned by more than ±12 steps, then the rod would be returned to ±12 steps or additional action must be taken as prescribed in the Technical Specification.

It is recognized that during certain reactor conditions the actual rod position cannot be determined. For example, during startup (subcritical) when the shutdown banks are withdrawn there may be misalignment, but because the reactor is subcritical, no independent verification possible. Therefore, the operator must rely on the RPI's. But, on the other hand, because there is no power, rod misalignment is of no significance. Therefore, the ±24 steps criteria for the RPI indication, when applied to actual rod misalignment would have no affect on thermal margins because of higher peaking factors. No increase in power is allowed until all shutdown banks are out, control bank A is out and control Banks B, C, and D are at or above the insertion limit.

Another situation where the actual rod position cannot be determined is when the reactor is being shutdown. Again for the control rods to be inserted beyond the insertion limit requires that the reactor be brought subcritical and again, rod misalignment would have no effect on thermal margins.

If it is determined that the RPI is out of calibration, on-line calibration of the instrumentation can be performed at or below the reduced power level given in the COLR but no higher than a nominal 30% power not to exceed 35% power. Thermal margins are maintained by reducing power to or below the respective COLR values for extended RPI deviation limits and on-line calibration.

If the rod position indicator channel is not operable, the operator will be fully aware of the inoperability of the channel, and special surveillance of core power tilt indications, using established procedures and relying on excore nuclear detectors and/or movable incore detectors, will be used to verify power distribution symmetry. These indirect measurements do not have the same resolution if the bank is near either end of the core, because a 24-step misalignment would have no significant effect on power distribution. Therefore, it is necessary to apply the indirect checks following significant rod motion.

One inoperable control rod is acceptable provided that the power distribution limits are met, trip shutdown capability is available, and provided the potential hypothetical

ejection of the inoperable rod is not worse than the cases analyzed in the safety analysis report. The rod ejection accident for an isolated fully-inserted rod will be worse if the residence time of the rod is long enough to cause significant non-uniform fuel depletion. The 4 week period is short compared with the time interval required to achieve a significant non-uniform
fuel depletion.

The required drop time to dashpot entry is consistent with safety analysis.

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Vice President-Nuclear Power shall be responsible for overall facility activities and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The General Manager-Nuclear Power Generation shall be responsible for facility operations and shall delegate in writing the succession to this responsibility during his absence.

6.2 ORGANIZATION

6.2.1 Facility Management and Technical Support

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the Updated FSAR.

b. The General Manager-Nuclear Power Generation shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. The Vice President-Nuclear Power shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

d. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

6.2.2 Facility Staff

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

b. At least one licensed Operator shall be in the control room when fuel is in the reactor.

c. At least two licensed Operators shall be present in the control room during reactor startup, scheduled reactor shutdown, and during recovery from reactor trips.

d. An individual qualified in radiation protection procedures shall be onsite when fuel is in the reactor.

e. All core alterations after the initial fuel loading shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling. This individual shall have no other concurrent responsibilities during this operation.

f. DELETED

g. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions (e.g., licensed Senior Operators, licensed Operators, health physicists, auxiliary operators, and key maintenance personnel).

The amount of overtime worked by unit staff members performing safety-related functions shall be limited in accordance with the NRC Policy Statement on working hours (Generic Letter No. 82-12).

h. The Operations Manager shall hold a senior reactor operator license.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for the Radiation Protection Manager who shall meet or exceed the minimum qualifications of Regulatory Guide 1.8, September 1975.

6.3.2 The General Manager-Nuclear Power Generation shall meet or exceed the minimum qualifications specified for Plant Manager in ANSI N18.1-1971.

6.3.3 The Watch Engineer shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Nuclear Training Director and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix A of 10 CFR Part 55.

6.4.2 DELETED

6.5 REVIEW AND AUDIT

6.5.1 Station Nuclear Safety Committee (SNSC)

Function

6.5.1.1 The Station Nuclear Safety Committee shall function to advise the Vice President-Nuclear Power on all matters related to nuclear safety.

Composition

6.5.1.2 The Station Nuclear Safety Committee shall, as a minimum, be composed as follows:

Chairman:	senior manager *
Member:	Chief Plant Engineer
Member:	Operations Manager
Member:	Maintenance Manager
Member:	Instrument and Control Engineer
Member:	Radiation Protection Manager
Member:	Reactor Engineer

* This senior manager shall be a technically competent person experienced in the field of nuclear energy, shall be appointed by and report directly to the Vice President-Nuclear Power for the SNSC function and shall be independent of the Nuclear Power Generation Organization.

6.5.1.2.1 In addition, other technically competent individuals may be appointed by the SNSC Chairman to serve as SNSC members.

Alternates

6.5.1.3 Alternate members shall be appointed in writing by the SNSC Chairman to serve on a temporary basis, and must have qualifications similar to the member being replaced.

Meeting Frequency

6.5.1.4 The SNSC shall meet at least once per calendar month and as convened by the SNSC Chairman or his designated alternate.

Quorum

6.5.1.5 A quorum of the SNSC shall consist of the Chairman or his designated alternate and four members. No more than two alternate members shall be included in the quorum.

Responsibilities

6.5.1.6 The Station Nuclear Safety Committee shall be responsible for:

a. review of (1) all procedures required by Specification 6.8 and changes thereto, and (2) any other proposed procedures or changes thereto as determined by the Chairman of SNSC to affect nuclear safety,

b. review of all proposed tests and experiments that affect nuclear safety,

c. review of all proposed changes to the Technical Specifications,

d. review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety,

e. investigation of all violations of the Technical Specifications and preparation and forwarding of a report covering evaluation and recommendations to prevent recurrence to the Vice President-Nuclear Power and to the Chairman of the Nuclear Facilities Safety Committee,

f. review of facility operations to detect potential nuclear safety hazards,

g. performance of special reviews and investigations and the issuance of reports thereon as required by the Chairman of the Nuclear Facilities Safety Committee,

h. review of the Plant Security Plan and implementing procedures and submission of recommended changes to the Chairman of the Nuclear Facilities Safety Committee,

i. review of the Emergency Plan and implementing procedures and submission of recommended changes to the Chairman of the Nuclear Facilities Safety Committee,

j. review of any unplanned, radioactive release, including the preparation of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Vice President-Nuclear Power and to the Nuclear Facility Safety Committee, and

k. review of changes to the Process Control Program and the Offsite Dose Calculation Manual,

l. review of the Fire Protection Program and implementing procedures and submission of recommended changes to the Chairman of the Nuclear Facilities Safety Committee.

Authority

6.5.1.7 The Station Nuclear Safety Committee shall:

a. recommend to the Vice President-Nuclear Power, in writing, approval or disapproval of items considered under Specifications 6.5.1.6(a) through (d) above,

b. render determinations, in writing, with regard to whether or not each item considered under Specifications 6.5.1.6(a) through (e) above constitutes an unreviewed safety question, and

c. provide immediate written notification to the Chairman, Nuclear Facilities Safety Committee of disagreement between the recommendations of the SNSC and the actions contemplated onsite. However, the course of action determined by the Vice President-Nuclear Power pursuant to Specification 6.1.1 above or the General Manager-Nuclear Power Generation pursuant to Specification 6.1.2 above shall be followed.

Records

6.5.1.8 The Station Nuclear Safety Committee shall maintain written minutes of each meeting and copies shall be provided to, as a minimum, the Vice President-Nuclear Power and the Chairman, Nuclear Facilities Safety Committee.

6.5.2 Nuclear Facilities Safety Committee (NFSC)

Function

6.5.2.1 The Nuclear Facilities Safety Committee shall function to provide independent review and audit of designated activities in the areas of:

a. reactor operations
b. nuclear engineering
c. chemistry and radiochemistry
d. metallurgy and non-destructive testing
e. instrumentation and control
f. radiological safety
g. mechanical and electrical engineering
h. administrative controls and quality assurance practices
i. radiological environmental effects
j. other appropriate fields associated with the unique characteristics of the nuclear power plant

Composition

6.5.2.2 The Committee shall have a permanent voting membership of at least 5 persons of which a majority are independent of the Nuclear Power organization and shall include technically competent persons from departments of Consolidated Edison having a direct interest in nuclear plant design, construction, operation or in nuclear safety. In addition, persons from departments not having a direct interest in nuclear plant design, construction, operation or nuclear safety may serve as members of the Committee if experienced in the field of nuclear energy. The Chairman and Vice Chairman will be senior officials of the Company experienced in the field of nuclear energy.

The Chairman of the Nuclear Facilities Safety Committee, hereafter referred to as the Chairman, shall be appointed by the Executive Vice President, Central Operations.

The Vice Chairman shall be appointed by the Executive Vice President, Central Operations. In the absence of the Chairman, he will serve as Chairman.

The Secretary shall be appointed by the Chairman of the Committee.

Committee members from departments having a direct interest in nuclear plant design, construction and operation or in nuclear safety shall be

designated by the Vice President of the Company, who is responsible for the functioning of the department subject to the approval of the Chairman. Committee members from other departments may be appointed by the Chairman with the concurrence of the Vice President of that department.

Alternates

6.5.2.3 Each permanent voting member, subject to the Chairman's approval, may appoint an alternate to serve in his absence. Committee records shall be maintained showing each such current designation.

No more than two alternates shall participate in activities at any one time.

Alternate members shall have voting rights.

Consultants

6.5.2.4 Consultants shall be utilized as determined by the NFSC Chairman.

Meeting Frequency

6.5.2.5 The NFSC shall meet at least once per calendar quarter or at more frequent intervals at the call of the Chairman or, in his absence, the Vice Chairman.

Quorum

6.5.2.6 A majority of the permanent voting committee members, or duly appointed alternates, which shall include the Chairman or the Vice Chairman and of which a minority are from the Nuclear Power Organization shall constitute a quorum for meetings of the Committee. In the event both the Chairman and the Vice Chairman are absent, one of the permanent voting members will serve as Acting Chairman.

Review

6.5.2.7 The following subjects shall be reported to and reviewed by the Committee insofar as they relate to matters of nuclear safety:

a. The safety evaluations for (1) changes to procedures, equipment or systems and (2) tests or experiments completed under the provision of 10

CFR 50.59 to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in 10 CFR 50.59.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59.

d. Proposed changes in Technical Specifications or licenses.

e. Violations of applicable statutes, codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.

g. Reportable Events, as specified by 10 CFR 50.73.

h. Any indication of an unanticipated deficiency in some aspect of design or operation of safety-related structures, systems, or components.

i. Reports and meeting minutes of the Station Nuclear Safety Committee.

j. Environmental surveillance program pertaining to radiological matters.

Audits

6.5.2.8 Audits of facility activities shall be performed under the cognizance of the NFSC. These audits shall encompass:

a. The conformance of facility operation to all provisions contained within the Radiological Technical Specifications (Appendix A) and applicable license conditions at least once per 12 months.

b. The conformance to all provisions contained within the Environmental Technical Specifications (Appendix B) pertaining to radiological matters and applicable license conditions at least once per 12 months.

c. The performance, training and qualifications of the entire facility staff at least once per 12 months.

d. The results of all actions taken to correct deficiencies occurring in facility equipment, structures, systems or method of operation that affect nuclear safety at least once per 6 months.

e. The performance of all activities required by the Quality Assurance Program to meet the criteria of Appendix B, 10 CFR 50, at least once per 24 months.

f. The Facility Emergency Plan and implementing procedures at least once per 12 months.

g. The Facility Security Plan and implementing procedures at least once per 12 months.

h. The Facility Fire Protection Program and implementing procedures at least once per 24 months.

i. A fire protection and loss prevention inspection and audit shall be performed utilizing either qualified offsite licensee personnel or an outside fire protection firm at least once per 12 months.

j. An inspection and audit of the fire protection and loss prevention program shall be performed by an outside qualified fire consultant at least once per 36 months.

k. The radiological environmental monitoring program and the results thereof at least once per 12 months.

l. The Offsite Dose Calculations Manual and implementing procedures at least once per 24 months.

m. The Process Control Program and implementing procedures for processing and packaging of radioactive wastes at least once per 24 months.

n. The performance of activities required by the Quality Assurance Program to meet the provisions of Regulatory Guide 1.21, Revision 1, June 1974 and Regulatory Guide 4.1, Revision 1, April 1975 at least once per 12 months.

o. Any other area of facility operation considered appropriate by the NFSC or the Executive Vice President, Central Operations.

Authority

6.5.2.9 The NFSC shall report to and advise the Executive Vice President, Central Operations on those areas of responsibility in Specifications 6.5.2.7 and 6.5.2.8.

Records

6.5.2.10 Records of NFSC activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each NFSC meeting shall be prepared, approved and forwarded to the Executive Vice President, Central Operations and to Senior Company Officers concerned with nuclear facilities within 14 days following each meeting.

b. Reports of reviews encompassed by Specifications 6.5.2.7 e, f, g and h above, shall be prepared, approved and forwarded to the Executive Vice President, Central Operations and to Senior Company Officers concerned with nuclear facilities within 14 days following completion of the review.

c. Audit reports encompassed by Specification 6.5.2.8 above, shall be forwarded to the Senior Company Officers concerned with nuclear facilities and to the management positions responsible for the areas audited within 30 days after completion of the audit.

6.6 REPORTABLE EVENT ACTION

6.6.0 A Reportable Event is defined as any of the conditions specified in 10 CFR 50.73a(2).

6.6.1 The following actions shall be taken in the event of a Reportable Event:

a. A report shall be submitted to the Commission pursuant to the requirements of 10 CFR 50.73.

b. Each Licensee Event Report submitted to the Commission shall be submitted to the NFSC Chairman and the Vice President-Nuclear Power and be reviewed by the SNSC.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The provisions of 10 CFR 50.36(c)(1)(i) shall be complied with immediately.

b. The Safety Limit Violation Report shall be reported to the Commission, the Vice President-Nuclear Power and to the NFSC Chairman immediately.

c. The Safety Limit Violation Report shall be prepared. The report shall be reviewed by the SNSC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, the NFSC Chairman and the Vice President-Nuclear Power within 10 days of the violation.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures and administrative policies shall be established, implemented and maintained covering the activities referenced below:

a. The requirements and recommendations of Sections 5.1 and 5.3 of ANSI N18.7-1972 and Appendix A of USAEC Regulatory Guide 1.33 (issued November 1972) except as provided in 6.8.2 and 6.8.3 below.

b. Process Control Program implementation.

c. Offsite Dose Calculation Manual implementation.

d. Quality Assurance Program for effluent and environmental monitoring using the guidance in Regulatory Guide 1.21, Revision 1, April 1974 and Regulatory Guide 4.1, Revision 1, April 1975.

e. Fire Protection Calculation Manual implementation.

6.8.2 Each procedure and administrative policy of Specification 6.8.1 above, and any changes to them shall be reviewed and approved for implementation in accordance with a written administrative control procedure approved by the

appropriate General Manager, with the concurrence of the Station Nuclear Safety Committee and the Vice President, Nuclear Power. The administrative control procedure required by this specification shall, as a minimum, require that:

a. Each proposed procedure/procedure change involving safety-related components and/or operation of same receives a pre-implementation review by the SNSC except in case of an emergency.

b. Each proposed procedure/procedure change which renders or may render the Updated Final Safety Analysis Report or subsequent safety analysis reports inaccurate and those which involve or may involve potential unreviewed safety questions are approved by the SNSC prior to implementation.

c. The approval of the Nuclear Facilities Safety Committee shall be sought if, following its review, the Station Nuclear Safety Committee finds that the proposed procedure/procedure change either involves an unreviewed safety question or if it is in doubt as to whether or not an unreviewed safety question is involved.

6.8.3 A mechanism shall exist for making temporary changes and they shall only be made by approved management personnel in accordance with the requirements of ANSI 18.7-1972. The change shall be documented, and reviewed by the SNSC and approved by a General Manager within 14 days of implementation.

6.8.4 The following programs shall be established, implemented, and maintained:

a. A program which will ensure the capability to obtain and analyze samples of reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere under accident conditions. The program shall include the following:

(i) training of personnel,

(ii) procedures for sampling and analysis, and

(iii) provisions for maintenance of sampling and analysis equipment.

6.9 REPORTING REQUIREMENTS

Routine Reports and Reportable Occurrences

6.9.1. In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator, Region I unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) amendments to the license involving a planned increase in power level, (2) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (3) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the appropriate tests identified in the UFSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.2 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

ANNUAL RADIATION EXPOSURE REPORT[1]

6.9.1.3 Routine reports of occupational radiation exposure data during the previous calendar year shall be submitted no later than March 1 of each year.

6.9.1.4 The annual radiation exposure reports shall provide a tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their

associated man rem exposure according to work and job functions[2], e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter TLD or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT[3]

6.9.1.5 Routine Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year.

The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and statistical evaluation of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, with operational controls as appropriate, and with previous environmental surveillance reports, and an assessment of the observed impacts of the plant operation on the environment. The report shall also include the results of land use censuses required by Specification 4.11.B.

The Annual Radiological Environmental Operating Reports shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements as described in the ODCM. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps[4] covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 4.11.C; discussion and all deviations from the sampling schedule

of Table 4.11-1; and discussion of all analyses in which the LLD required by Table 4.11-3 was not achievable.

RADIOACTIVE EFFLUENT REPORT[5]

6.9.1.6 Routine Radioactive Effluent Release Reports covering the operation of the unit during the previous 12 months of operation shall be submitted by May 1 of each year.

The Radioactive Effluent Release Report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

The Radioactive Effluent Release Report to be submitted by May 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distribution of wind speed, wind direction, and atmospheric stability[6].

This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents releases from the unit or station during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to members of the public due to their activities inside the site boundary (Figure 5.1-1) during the report period. All assumptions used in making these assessments, i.e., specific activity, exposure time and location, shall be included in these reports. The meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents, as determined by sampling frequency and measurement, shall be used for determining the gaseous pathway doses. Approximate and conservative approximate methods are acceptable. The assessment of radiation doses shall be performed in accordance with the methodology and parameters in the Offsite Dose Calculation Manual (ODCM).

Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1, October 1977.

The Radioactive Effluent Release Report shall include the following information for each class of solid waste (in compliance with 10 CFR Part 61) shipped offsite during the report period:

a. container volume,

b. total curie quantity (specify whether determined by measurement or estimate),

c. principal radionuclides (specify whether determined by measurement or estimate),

d. source of waste and processing employed (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms),

e. type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f. solidification agent or absorbent (e.g., cement, urea formaldehyde).

The Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to Unrestricted Areas of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Radioactive Effluent Release Report shall include any changes made during the reporting period to the Process Control Program (PCP) and to the Offsite Dose Calculation Manual (ODCM), as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 4.11.B.

MONTHLY OPERATING REPORT

6.9.1.7 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or pressurizer safety valves shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, no later than the 15th of each month following the calendar month covered by the report.

6.9.1.8 Core operating limits shall be established and documented prior to each reload cycle, or prior to any remaining portion of the cycle, for the following:

a. Axial Flux Difference limits for Specifications 3.10.2.

b. Height Dependent Heat Flux Hot Channel Factor for Specification 3.10.2.

c. Nuclear Enthalpy Rise Hot Channel Factor for Specification 3.10.2.

d. Shutdown Bank Insertion Limit for Specification 3.10.4.

e. Control Bank Insertion Limits for Specification 3.10.4.

6.9.1.9 The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

a. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary). (Methodology for Specification 3.10.4 - Shutdown Bank Insertion Limit, Control Bank Insertion Limits and 3.10.2 - Nuclear Enthalpy Rise Hot Channel Factor.)

b. WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES - TOPICAL REPORT", September 1974 (W Proprietary). (Methodology for Specification 3.10.2 - Axial Flux Difference (Constant Axial Offset Control).)

c. T.M. Anderson to K. Kniel (Chief of Core Performance Branch, NRC) January 31, 1980 - Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package. (Methodology for Specification 3.10.2 - Axial Flux Difference (Constant Axial Offset Control).)

d. NUREG-0800, Standard Review Plan, US Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981. Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981. (Methodology for Specification 3.10.2 - Axial Flux Difference (Constant Axial Offset Control).)

e. WCAP-10266-P-A Rev. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE", March 1987, (W Proprietary). (Methodology for Specification 3.10.2 Height Dependent Heat Flux Hot Channel Factor.)

f. WCAP-12945-P, Westinghouse "Code Qualification Document for Best Estimate LOCA Analyses", July, 1996

6.9.1.10 The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling System (ECCS) limits, nuclear limits such as shutdown margin, transient analysis limits, and accident analysis limits) of the safety analysis are met.

6.9.1.11 The COLR, including any mid-cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

Special Reports

6.9.2 Special reports shall be submitted to the NRC Regional Administrator of the Region I Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. Each containment integrated leak rate test shall be the subject of a summary technical report including results of the local leak rate test since the last report. The report shall include analyses and interpretations of the results which demonstrate compliance in meeting the leak rate limits specified in the Technical Specifications.

b. DELETED

c. Sealed source leakage in excess of limits (Specification 4.15).

d. The complete results of the steam generator tube inservice inspection (Specification 4.13.C.).

e. Radioactive effluents (Specification 3.9).

f. Radiological environmental monitoring (Specification 4.11).

g. Meteorological monitoring instrumentation (Specification 3.15).

h. Inoperable radiation and hydrogen monitoring instrumentation (Specification 3.5) outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to operable status.

i. Operation of overpressure protection system (Specification 3.1.A.4).

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time intervals at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. Reportable Event Reports.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of reactor tests and experiments.

f. Records of changes made to Operating Procedures.

g. Records of radioactive shipments.

h. Records of sealed source leak tests and results.

i. Records of annual physical inventory of all source material on record.

6.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Record and drawing changes reflecting facility design modifications made to systems and equipment described in the Updated Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of facility radiation and contamination surveys.

d. Records of radiation exposure for all individuals entering radiation control areas.

e. Records of gaseous and liquid radioactive material releases to the environs.

f. Records of transient or operational cycles for those facility components designed for a limited number of transients or cycles.

g. Records of training and qualification for current members of the plant staff.

h. Records of inservice inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities required by the QA Manual except as noted in 6.10.1.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the SNSC and the NFSC.

l. Records for Environmental Qualification which are covered under the provisions of Specification 6.13.

m. Records of analyses required by the radiological environmental monitoring program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed.

n. Records of the service lives of all snubbers addressed by Section 3.12 of the Technical Specifications, including the date at which the service life commences and associated installation and maintenance records.*

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 As an acceptable alternative to the "control device" or "alarm signal" required by 10 CFR 20.203(c)(2):

a. Each High Radiation Area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a High Radiation Area and entrance thereto shall be controlled by issuance of a Radiation Work Permit and any individual or group of individuals permitted to enter such areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. Each High Radiation Area in which the intensity of radiation is greater than 1000 mrem/hr shall be subject to the provisions of Specification 6.12.1(a) above, and in addition locked doors shall be provided to prevent unauthorized entry to such areas and the keys shall be maintained under the administrative control of the Radiation Protection Manager and/or the Senior Watch Supervisor on duty.

6.13 ENVIRONMENTAL QUALIFICATION

6.13.1 By no later than June 30, 1982 all safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of Division of Operating Reactors "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors" (DOR Guidelines), or NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," December 1979. Copies of these documents are attached to Order for Modification of License No. DPR-26 dated October 24, 1980.

6.13.2 By no later than December 1, 1980, complete and auditable records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical

equipment in sufficient detail to document the degree of compliance with the DOR Guidelines of NUREG-0588. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.

6.14 PROCESS CONTROL PROGRAM (PCP)

6.14.1 Licensee initiated changes to the PCP:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

   a. sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information,

   b. a determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes, and

   c. documentation of the fact that the change has been reviewed and found acceptable by the SNSC.

2. Shall become effective upon review and acceptance by the SNSC.

6.15 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.15.1 The ODCM shall be approved by the Commission prior to implementation.

6.15.2 Licensee initiated changes to the ODCM:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

   a. sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of those pages of the ODCM to be changed with each page numbered and

provided with an approval and date box, together with appropriate analyses or evaluation justifying the change(s),

b. a determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations, and

c. documentation of the fact the change has been revised and found acceptable by the SNSC.

2. Shall become effective upon review and acceptance by the SNSC.

6.16 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS AND SOLID WASTE SYSTEMS

6.16.1 Licensee initiated major changes to the radioactive waste systems (liquid, gaseous and solid) shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change was made. The discussion of each change shall contain:

a. a summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR Part 50.59,

b. sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information,

c. a detailed description of the equipment, components and processes involved and the interfaces with other plant systems,

d. an evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto,

e. an evaluation of the change, which shows the expected maximum exposures to individuals in the Unrestricted Area and to the general population that differ from those previously estimated in the license application and amendments thereto,

f. a comparison of the predicted releases of radioactive materials in liquid and gaseous effluents and in solid waste to the actual releases for the period in which the changes are to be made;

g. an estimate of the exposure to plant operating personnel as a result of the change, and

h. documentation of the fact that the change was reviewed and found acceptable by the SNSC.

---

* The documentation referred to herein is required for all snubbers beginning with those replaced following the issuance of Amendment 112.

1 A single submittal may be made for a multiple-unit station. The submittal should combine those sections that are common to all units at the station.

2 This tabulation supplements the requirements of 10 CFR Part 20.407.

3 A single submittal may be made for a multiple unit station.

4 One map shall cover stations near the site boundary; a second shall include more distant stations.

5 A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

6 In lieu of submission with the Radioactive Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data onsite in a file that shall be provided to the NRC upon request.

ATTACHMENT II

SAFETY ASSESSMENT

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket 50-247
August, 1996

SECTION I - Description of Changes

This application for amendment to the Con Edison Indian Point Unit No. 2 Technical Specifications seeks to amend Section 3.3 (Engineered Safety Features, the basis of Section 3.3, the basis of Section 3.6 (Containment), the basis of Section 3.10 (Control Rods) and Section 6.9.1.9 (Core Operating Limits Report (COLR)).

These changes are being made to incorporate the best estimate approach into the licensing basis for the Indian Point Unit No. 2 large break LOCA analyses in accordance with 10 CFR 50.46, Regulatory Guide (number), Westinghouse "Code Qualification Document For Best Estimate LOCA Analysis," WCAP-12945-P, Volumes I-V and NRC Safety Evaluation Report [number] [date]; and to revise the Technical Specification limits for several plant parameters used in the analysis. UFSAR changes will be made in accordance with 50.71(e).

The proposed changes are specified in Attachment I to the Application for Amendment enclosed with this letter.

Table 1 lists the plant specific parameters used in the Indian Point Unit No. 2 plant specific analysis and the location of the documentation of the values and ranges used for the parameters.

Table 2 presents the 50th and 95th percentile Peak Clad Temperature (PCT) for Indian Point Unit No. 2, maximum cladding oxidation, maximum hydrogen generation, and cooling results.

SECTION II - Evaluation of Changes

A best estimate large break loss of coolant accident (LOCA) analysis has been performed for Indian Point Unit No. 2. The analysis is contained in "Best Estimate Analysis of the Large Break LOCA Analysis For Indian Point Unit 2", EPRI-TR-103391/WCAP-13837. The approved Westinghouse best estimate methodology contained in WCAP-12954-P was used. All plant specific parameters used in the analysis are bounded by the models and correlations contained in the generic methodology. Therefore, the Indian Point Unit No. 2 specific analysis conforms to 10 CFR 50.46 and Appendix K and meets the intent of Regulatory Guide 1.157.

The conclusions of the analysis are that there is a high level of probability that:

1) The calculated maximum fuel element cladding temperature (peak cladding temperature) will not exceed 2200°F.

2) The calculated total oxidation of the cladding (maximum cladding oxidation) will nowhere exceed 0.17 times the total cladding thickness before oxidation.

3) The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam (maximum hydrogen generation) will not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

4) The calculated changes in core geometry are such that the core remains amenable to cooling.

and 5) After successful initial operation of the ECCS, the core temperature will be maintained at an acceptably low value and decay heat will be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

SECTION III - No Significant Hazards Evaluation

Consistent with the requirements of 10 CFR 50.92, the enclosed application involves no significant hazards based on the following information:

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

Response:

No physical changes are being made by this change. The plant conditions assumed in the analysis are bounded by the design conditions for all equipment in the plant. Therefore, there will be no increase in the probability of a loss of coolant accident. The consequences of a LOCA are not being increased. That is, it is shown that the emergency core cooling system is designed so that its calculated cooling performance conforms to the criteria contained in 50.46 paragraph b, that is it meets the five criteria listed in Section II of this evaluation. No other accident is potentially affected by this change. Therefore, neither the probability nor the consequences of an accident previously analyzed is increased due to the proposed change.

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

Response:

There are no physical changes being made to the plant. No new modes of plant operation are being introduced. The parameters assumed in the analysis are within the design limits of existing plant equipment. All plant systems will perform equally during the response to a potential accident. Therefore, the possibility of a new or different kind of accident than previously analyzed will not be increased.

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

Response:

It has been shown that the analytic technique used in the analysis realistically describes the expected behavior of the Indian Point Unit No. 2 reactor system during a postulated loss of coolant accident. Uncertainties have been accounted for as required by 10 CFR 50.46. A sufficient number of loss of coolant accidents with different break sizes, different locations and other variations in properties have been analyzed to provide assurance that the most severe postulated loss of coolant accidents were calculated. It has been shown by the analysis that there is a high level of probability that all criteria contained in 10 CFR 50.46 paragraph b) are met. Therefore the proposed amendment does not involve a significant reduction in the margin of safety.

SECTION IV - Impact of Changes

These changes will not adversely impact the following:

ALARA Program
Security and Fire Protection Programs
Emergency Plan
FSAR of SER Conclusions
Overall Plant Operations and the Environment

SECTION V - Conclusions

The incorporation of these changes: a) will not increase the probability or the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Safety Analysis Report; b) will not create the possibility for an accident or malfunction of a new or different kind form any evaluated previously in the Safety Analysis Report; c) will not reduce the margin of safety as defined in the bases for any Technical Specification; d) dos not constitute an unreviewed safety question; and e) involves no significant hazards considerations as defined in 10 CFR 50.92.

The proposed changes have been reviewed by the Station Nuclear Facility Committee and the Nuclear Facility Safety Committee.