APPENDIX A

TO

FACILITY OPERATING LICENSE DPR-26

FOR

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

DOCKET NO. 50-247

TECHNICAL SPECIFICATIONS

FOR

FUEL LOADING AND SUBCRITICAL TESTING

NOTE: These Technical Specifications consist of portions of the proposed Technical Specifications as submitted with Amendment 25 (Supplement 15) on November 12, 1970 with appropriate additions directed to the operations of fuel loading and subcritical testing. Paragraph and page numbering are therefore not necessarily in sequential order but are as given in the Table of Contents.

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# 1 DEFINITIONS

The following used terms are defined for uniform interpretation of the specifications.

## 1.1 Reactor Operating Conditions

### 1.1.1 Cold Shutdown Condition

When the reactor is subcritical by at least 1% $\Delta$ k/k and $T_{avg}$ is $\leq$ 140° F.

### 1.1.2 Refueling Operating Condition

Any Operation involving movement of core components when the vessel head is completely unbolted.

## 1.2 Operable

A system or component is operable when it is capable of performing its intended function within the required range.

## 1.3 Instrumentation Surveillance

### 1.3.1 Channel Check

A qualitative determination of acceptable operability by observation of channel behavior during operation. This determination shall include comparison of the channel with other independent channels measuring the same variable.

### 1.3.2 Channel Calibration

Adjustment of channel output such that it responds, with acceptable range and accuracy, to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including alarm.

## 1.4 Abnormal Occurrence

An abnormal occurrence means the occurrence of any plant condition that:

a. Results in a protective instrumentation setting in excess of a Limiting Safety System Setting as established in the Technical Specifications, or

b. Exceeds a Limiting Condition for Operation as established in the Technical Specifications, or

c. Causes any release of radioactivity from the site, or

d. Results in uncontrolled or unanticipated changes in reactivity greater than 1% $\Delta$ k/k.

1.5 <u>Locked valves</u>

A manually operated valve is considered to be locked closed if it is in the closed position and there is a chain through the handwheel secured with a padlock. A motor operated valve is considered to be locked closed if it is in the closed position with the power disconnected at the motor control center and if the handwheel (if any) is chained and padlocked as above.

1.7 <u>Containment Integrity</u>

Containment integrity is defined to exist when:

a. The equipment door and at least one door in each personnel air lock is properly closed.

b. At least one isolation valve is operable or locked closed in each line penetrating containment and which provides a direct path from the containment atmosphere to the outside.

## 2.2 SAFETY LIMIT: REACTOR COOLANT SYSTEM PRESSURE

Applicability

Applies to the maximum limit on Reactor Coolant System pressure.

Objective

To maintain the integrity of the Reactor Coolant System and to prevent the release of excessive amounts of fission product activity to the containment.

Specification

The Reactor Coolant System pressure shall not exceed 2735 psig with fuel assemblies installed in the reactor vessel.

Basis

The Reactor Coolant System(1) serves as a barrier preventing radionuclides contained in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure the Reactor Coolant System is the primary barrier against the release of fission products. By establishing a system pressure limit, the continued integrity of the Reactor Coolant System is assured. The maximum transient pressure allowable in the Reactor Coolant System pressure vessel under the ASME Code, Section III is 110% of design pressure. The maximum transient pressure allowable in the Reactor Coolant System piping, valves and fittings under USAS Section B31.1 is 120% of design pressure. Thus, the safety limit of 2735 psig (110% of design pressure) has been established.

The setting of the power operated relief valves (2335 psig)[2] and the reactor high pressure trip (2385 psig)[2] have been established to assure that the Reactor Coolant System pressure limit is never reached and that the system pressure does not exceed the design limits of the fuel cladding.

In addition, the Reactor Coolant System safety valves[3] are sized to prevent system pressure from exceeding the design pressure by more than 10 percent (2735 psig) in accordance with Section III of the ASME Boiler and Pressure Vessel Code, assuming complete loss of load without a direct reactor trip or any other control, except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valves settings.

As an assurance of system integrity, all components in the system are hydrotested at 3110 psig prior to initial operation.[4]

References


(1) FSAR Section 4

(2) FSAR Table 4.1-1

(3) FSAR Section 4.3.4

(4) FSAR Section 4.3.3

## 3 LIMITING CONDITIONS FOR OPERATION

### 3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the Reactor Coolant System.

Objective

To specify those limiting conditions for operation of the Reactor Coolant System which must be met to ensure safe reactor operation.

Specification

A. OPERATIONAL COMPONENTS

1. Coolant Pumps

   a. At least one reactor coolant pump or one residual heat removal pump in the Residual Heat Removal System when connected to the Reactor Coolant System shall be in operation when a reduction is made in the boron concentration of the reactor coolant.

   b. When the reactor is critical and above 2% rated power, except for natural circulation tests, at least two reactor coolant pumps shall be in operation.

   c. Reactor power shall not be increased above 60% of rated power with only three pumps in operation unless the overtemperature

ΔT trip setpoint for three loop operation has been set in accordance with specification 2.3.1.B-4.

2. Steam Generator

Two steam generators shall be capable of performing their heat transfer function whenever the reactor is critical and the average coolant temperature is above 350°F.

3. Safety Valves

a. At least one pressurizer code safety valve shall be operable whenever the reactor head is on the vessel except for hydrostatically testing the RCS in accordance with the ASME Section XI Boiler and Pressure Vessel Code.

b. All pressurizer code safety valves shall be operable whenever the reactor is critical.

c. The pressurizer code safety valve lift setting shall be set at 2485 psig with ± 1% allowance for error.

Basis

When the boron concentration of the Reactor Coolant System is to be reduced the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the primary system volume in approximately one half hour. The pressurizer is of no concern because of the low pressurizer volume and because the pressurizer boron concentration will be higher than that of the rest of the reactor coolant.

Heat transfer analyses show that reactor heat equivalent to 10% of rated power can be removed with natural circulation only[1]; hence, the specified upper limit of 2% rated power without operating pumps provides a substantial safety factor.

Each of the pressurizer code safety valves is designed to relieve 408,000 lbs. per hr. of saturated steam at the valve set point. Below approximately 350°F and 450 psig in the Reactor Coolant System, the Residual Heat Removal System can remove decay heat and thereby control system temperature and pressure.[2]

If no residual heat were removed by the Residual Heat Removal System the amount of steam which could be generated at safety valve relief pressure would be less than half the capacity of a single valve. One valve therefore provides adequate protection for over-pressurization.

The combined capacity of the three pressurizer safety valves is greater than the maximum surge rate resulting from complete loss of load[3] without a direct reactor trip or any other control.

Two steam generators capable of performing their heat transfer function will provide sufficient heat removal capability to remove core decay heat after a reactor shutdown.

<u>Reference</u>

1) FSAR Section 14.1.6
2) FSAR Section 9.3.1
3) FSAR Section 14.1.10

B. HEATUP AND COOLDOWN

1. For the first two years of power operation (1.61 x $10^6$ thermal megawatt days) the reactor coolant pressure and the system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1-1 and Figure 3.1-2, and are as follows:

Heatup:

a. For indicated temperatures at or below 220°F the maximum indicated pressure shall not exceed 500 psig and the maximum heatup rate shall not exceed 50°F/hr, as shown by the dotted line on Figure 3.1-1.

b. For indicated temperatures above 220°F the heatup rate shall not exceed 100°F/hr.

Cooldown:

a. Allowable combinations of pressure and temperature for a specific cooldown rate for indicated temperature at or below 136°F are below and to the right of the solid limit lines for that rate as shown on Figure 3.1-2. Furthermore, the maximum indicated pressure shall not exceed 500 psig for indicated temperatures at or below 220°F as shown by the dotted limit line on Figure 3.1-2. The maximum cooldown rate shall not exceed 50°F/hr for indicated temperature at or below 220°F. The limit lines for cooling rates between those shown by the solid lines on Figure 3.1-2 may be obtained by interpolation.

b. For indicated temperatures above 220°F the rate shall not exceed 100°F/hr.

2. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the vessel is below 70°F.

3. Pressurizer

The heatup and cooldown rates shall not exceed 200°F/hr. The spray should not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

4. Within two years of power operation, Figures 3.1-1 and 3.1-2 shall be updated in accordance with appropriate criteria accepted by the AEC.

Basis

All components in the Reactor Coolant System are designed to withstand the effects of the cyclic loads due to reactor system temperature and pressure changes.[(1)] These cyclic loads are introduced by normal unit load transients, reactor trips, and startup and shutdown operation. The number of thermal and loading cycles used for design purposes are shown in Table 4.1-8 of the FSAR. During unit startup and shutdown, the rates of temperature and pressure changes are limited. The maximum plant heatup and cooldown rate of 100°F per hour is consistent with the design number of cycles and satisfies stress limits for cyclic operation.[(2)]

The reactor vessel plate opposite the core has been purchased to a specified Charpy V-notch test result of 30 ft-lb or greater at a nil-ductility transition temperature (NDTT) of 40°F or less. The material has been tested to verify conformity to specified requirements and a NDTT value of 20°F has been determined. In addition, this plate has been 100 percent volumetrically inspected by ultrasonic test using both longitudinal and shear wave methods. The remaining material in the reactor vessel, and other Reactor Coolant System components, meet the appropriate design code requirements and specific component function.[(3)]

As a result of fast neutron irradiation in the region of the core, there will be an increase in the NDTT, with nuclear operation. The techniques used to measure and predict the integrated fast neutron (E > 1 Mev) fluxes at the sample location are described in Appendix 4A of the FSAR. The calculation method used to obtain the maximum neutron (E > 1 Mev) exposure of the reactor vessel is identical to that described for the irradiation samples.

Since the neutron spectra at the samples and vessel inside radius are identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of reactor vessel for some later stage in plant life. The maximum exposure of the vessel will be obtained from the measured sample exposure by appropriate application of the calculated azimuthal neutron flux variation.

The maximum integrated fast neutron (E > 1 Mev) exposure of the vessel is computed to be $2.4 \times 10^{19}$ n/cm$^2$ for 40 years operation at 80 percent load factor.[3] The predicted NDTT shift for an integrated fast neutron (E > 1 Mev) exposure of $2.4 \times 10^{19}$ n/cm$^2$ is 238°F, the value obtained from the curve shown in Figure 4.2-9 of the FSAR for 550°F irradiation.[3]

The actual shift in NDTT will be established periodically during plant operation by testing vessel material samples which are irradiated cumulatively by securing them near the inside wall of the vessel in the core area. To compensate for any increase in the NDTT caused by irradiation, the limits on the pressure-temperature relationship are periodically changed to stay within the stress limits during heatup and cooldown.

During the first two years of reactor operation, a conservatively high estimate of the energy output is $1.61 \times 10^6$ thermal megawatt days, which is equivalent to 584 days at 2758 MWt. The projected fast neutron exposure of the vessel for this interval of operation is $1.2 \times 10^{18}$ n/cm$^2$ and the corresponding NDTT shift is 46°F, based on the curve shown in Figure 4.2-9 of the FSAR for 550°F irradiation. Thus, for this interval, the upper limit to the NDTT is 66°F. The corresponding Design Transition Temperature, defined as NDTT + 60°F,[4] is 126°F.

The stress allowed in the vessel in relation to operation below NDTT and DTT (NDTT + 60) to preclude the possibility of brittle failure are:

1. At DTT; a maximum stress of 20% yield
2. For DTT to DTT minus 200°F; a maximum stress decreasing from 20% to 10% yield
3. Below DTT minus 200°F; a maximum stress of 10% yield

These limits are based on the data reported by Kibara and Masubichi (Effect of Residual Stress on Brittle Fracture, April 1959, Welding Journal Volume 38) and Robertson (Propagation from Brittle Fracture in Steel, Journal of the Iron and Steel Institute, 1953), which show that if the stresses are maintained within the above limits, brittle fracture does not occur.[5]

The solid limit lines in Figure 3.1-2 are based on these stress limits and contain allowances for the 10°F margin between actual and measured temperature and 60 psi margin between actual and measured pressure.

During cooldown, the thermal stress varies from tensile at the inner wall to compressive at the outer wall. The internal pressure superimposes a tensile stress on this thermal stress pattern, increasing the stress at the inside wall and relieving the stress at the outside wall. Therefore the limiting stress always appears at the inside wall, so the limit line has a direct dependence on cooldown rate. This leads to a family of curves for cooldown, as shown by the solid lines on Figure 3.1-2.

For heatup, the thermal stress is reversed and the location of the limiting stress is a function of the heatup rate. The limit lines no longer bear the simple relationship to heatup as they do to cooldown rate. The limit lines based on the stress limits are not shown on Figure 3.1-1, since they are less restrictive than the limits described below and shown by the dotted line on Figure 3.1-1.

For additional conservatism in fracture toughness concepts including a size effect by the AEC Regulatory Staff, a maximum pressure of 560 psig below 210°F with a maximum heatup and cooldown rate of 50°F/hr was imposed for a two year period as shown by the dotted lines on Figures 3.1-1 and 3.1-2.

During this two year period, a fracture toughness criterion applicable to the operation of IPP plant beyond this period will be developed. It will be based on the evaluation of the fracture toughness properties of heavy section steels, both irradiated and unirradiated, for the AEC - HSST program and the PVRC program, and with consideration of test results of the IPP reactor surveillance program.

The NDTT shift and the magnitudes of the thermal and pressure stresses are sensitive to integrated reactor power and not to instantaneous power level. Figures 3.1-1 and 3.1-2 are applicable to thermal ratings up to 2758 MWt.

Figures 3.1-1 and 3.1-2 define stress limitations only. For normal operation other inherent plant characteristics, e.g., pump parameter or pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure ranges.

The heatup and cooldown rate of 100°F per hour for the steam generator is consistent with the remainder of the Reactor Coolant System, as discussed in the first paragraph of the Basis. The stresses are within acceptable limits for the anticipated usage. The spray should not be used if the temperature difference between the pressurizer and spray fluid is greater than 320°F. This limit is imposed to maintain the thermal stresses at the pressurizer spray line nozzle below the design limit. Temperature requirements for steam generator correspond with the measured NDT for the shell.

References:


(1) FSAR, Section 4.1.5

(2) ASME Boiler and Pressure Vessel Code, Section III, N-415

(3) FSAR, Section 4.2.5

(4) ASME Boiler and Pressure Vessel Code, Section III, N-331

(5) FSAR, Section 4.3.1

E. MAXIMUM REACTOR COOLANT OXYGEN, CHLORIDE AND FLUORIDE CONCENTRATION

Specification

1. Concentrations of contaminants in the reactor shall not exceed the following limits when the reactor coolant is above 250°F:

| Contaminant | Normal Steady-State Operation (PPM) | Transients not to Exceed 24 Hours (PPM) |
|---|---|---|
| a. Oxygen | 0.10 | 1.00 |
| b. Chloride | 0.15 | 1.50 |
| c. Fluoride | 0.15 | 1.50 |

2. If any of the normal steady-state operating limits as specified in 3.1.E.1 above are exceeded, or if it is anticipated that they may be exceeded, corrective action shall be taken immediately.

3. If the concentrations of any of the contaminants can not be controlled within the limits of Specification 3.1.E.1 above, the reactor shall be brought to the cold shutdown condition, utilizing normal operating procedures, and the cause of the out-of-specification operation ascertained and corrected. The reactor may then be restarted and operation resumed if the maximum concentration of any of the contaminants did not exceed the permitted transient values. Otherwise, a safety review is required before startup.

4. Concentrations of contaminants in the reactor coolant shall not exceed the following maximum limits when the reactor coolant temperature is below 250°F:

| Contaminant | Normal Concentration (PPM) | Transient not to exceed 48 hours (PPM) |
|---|---|---|
| a. Oxygen | Saturated | Saturated |
| b. Chloride | 0.15 | 1.5 |
| c. Fluoride | 0.15 | 1.5 |

If the limits above are exceeded, the reactor shall be immediately brought to the cold shutdown condition and the cause of the out-of-specification condition are ascertained and corrected.

5. For the purposes of correcting the contaminant concentrations to meet specifications 3.1.E.1 and 3.1.E.4 above, increase in coolant temperature consistant with operation of reactor coolant pumps for a short period of time to assure mixing of the coolant shall be permitted. This increase in temperature to assure mixing shall in no case cause the coolant temperature to exceed 250°F.

Basis:

By maintaining the oxygen, chloride and fluoride concentrations in the reactor coolant below the limits as specified in 3.1.E.1 and 3.1.E.4 the integrity of the reactor coolant system is assured under all operating conditions.(1)

If these limits are exceeded, measures can be taken to correct the condition, e.g., replacement of ion exchange resin or adjustment of the hydrogen concentration in the volume control tank(2), and further because of the time dependent nature of any adverse effects arising from oxygen, chloride, and fluoride concentration in excess of the limits, it is unnecessary to shut down immediately since the condition can be corrected. Thus the period of 24 hours for corrective action to restore concentrations within the limits has been established. If the corrective action has not been effective at the end of the 24 hour period, then the reactor will be brought to the cold shutdown condition and the corrective action will continue.

The effects of contaminants in the reactor coolant are temperature dependent. It is consistent, therefore, to permit a transient concentration to exist for a longer period of time and still provide the assurance that the integrity of the primary coolant system will be maintained.

In order to restore the contaminant concentrations to within specification limits in the event such limits were exceeded, mixing of the primary coolant with the reactor coolant pumps may be required. This will result in a small heatup of short duration and will not increase the average coolant temperature above 250°F.


References

(1) FSAR Section 4.2

(2) FSAR Section 9.2

REACTOR COOLANT SYSTEM
HEATUP LIMITATIONS

Applicable to 1.61 x $10^6$ MWt - Days

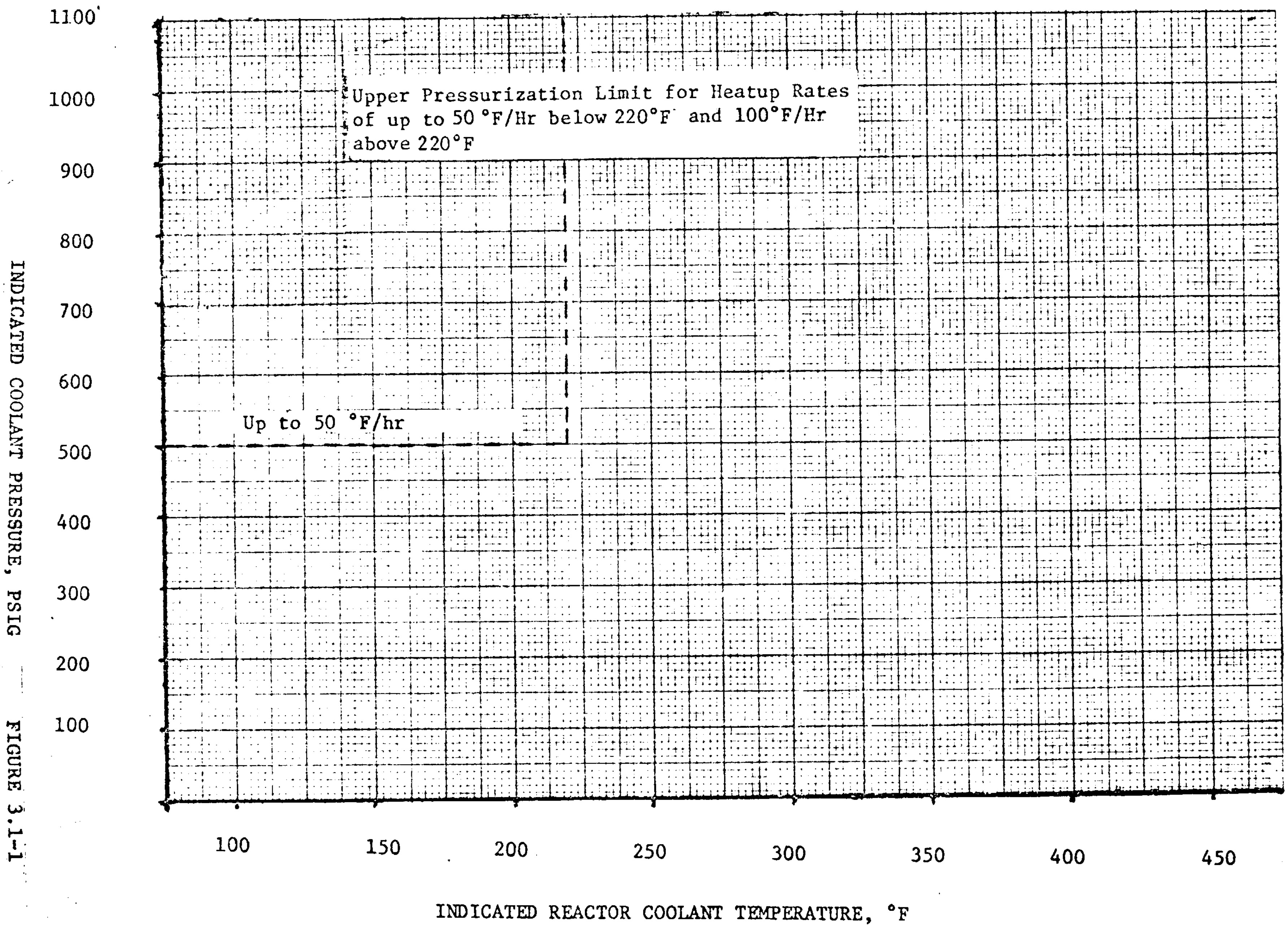



FIGURE 3.1-1

REACTOR COOLANT SYSTEM
COOLDOWN LIMITATIONS

Applicable to $1.61 \times 10^6$ $MW_t$ - Days

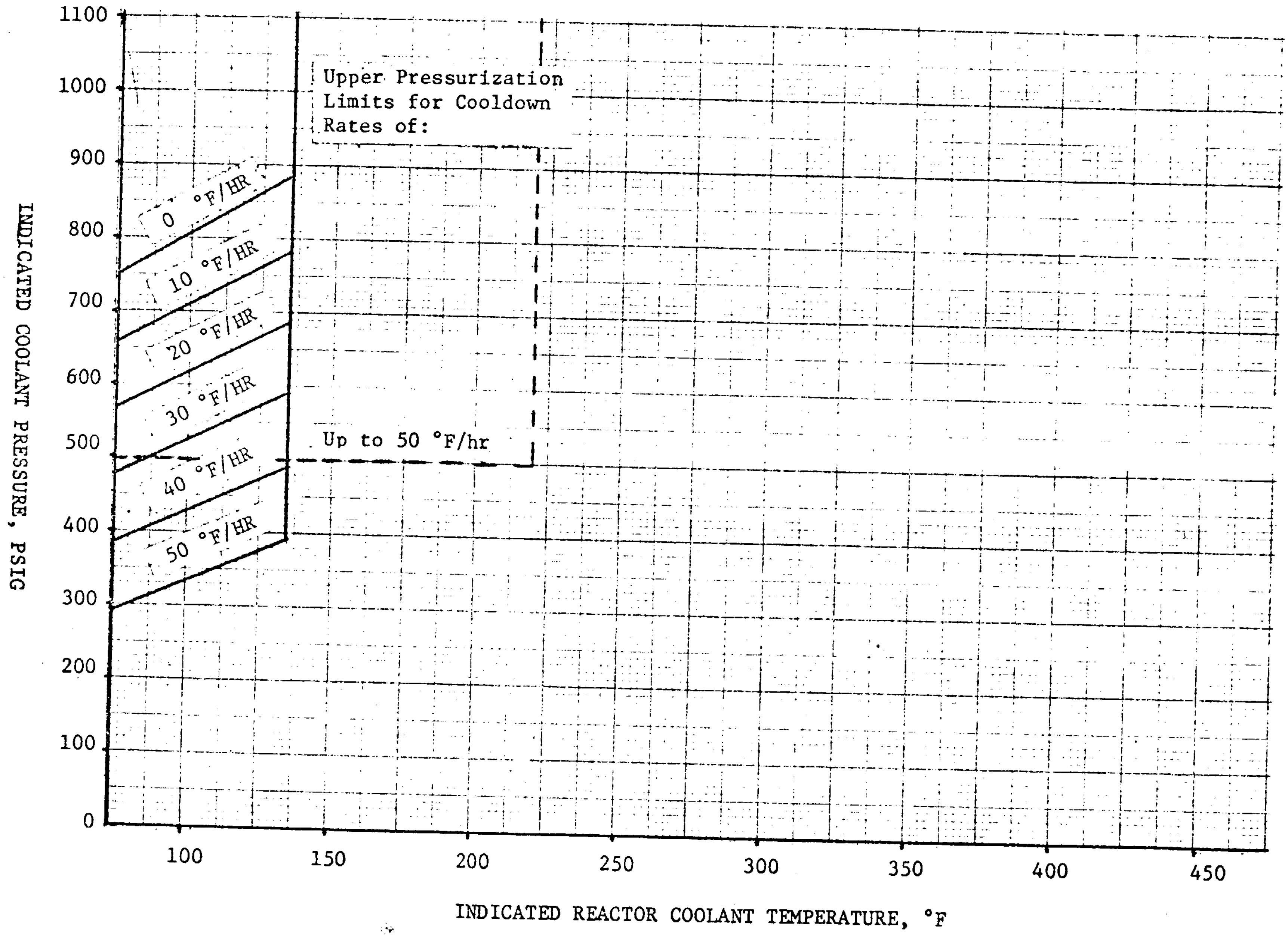



FIGURE 3.1-2

## 3.2 CHEMICAL AND VOLUME CONTROL SYSTEM

Applicability

Applies to the operational status of the Chemical and Volume Control System.

Objective

To define those conditions of the Chemical and Volume Control System necessary to ensure safe reactor operation.

Specification

A. When fuel is in the reactor there shall be at least one flow path to the core for boric acid injection.

B. The reactor shall not be made critical unless the following Chemical and Volume Control System conditions are met.

1. Two charging pumps shall be operable.

2. Two boric acid transfer pumps shall be operable.

3. The boric acid tanks together shall contain a minimum of 4400 gallons of 11 1/2% to 13% by weight (20,000 ppm to 25,500 ppm of boron) boric acid solution at a temperature of at least 145°F.

4. System piping and valves shall be operable to the extent of establishing one flow path from the boric acid tanks and one flow path from the refueling water storage tank to the Reactor Coolant System.

5. Two channels of heat tracing shall be operable for the flow path from the boric acid tanks.

C. During power operation, the requirements of 3.2.B 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 3.2.B within the time period specified, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If the requirements of 3.2.B are not satisfied within an additional 48 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operation procedures.

1. One of the two operable charging pumps may be removed from service provided a charging pump is restored to operable status within 24 hours.

2. One boric acid transfer pump may be out of service provided the pump is restored to operable status within 24 hours.

3. One boric acid tank may be out of service provided a minimum of 4400 gallons of 11 1/2% to 13% by weight (20,000 ppm to 22,500 ppm of boron) boric acid solution at a temperature of at least 145°F is contained in the operable tank and provided that the tank is restored to operable status within 48 hours.

4. One channel of heat tracing may be out of service for 48 hours.

D. Extended Maintenance

When it is determined that maintenance to restore components or systems to an operable condition will last longer than periods specified, the circumstances of the extended maintenance and the estimated date for returning the components or systems to an operable condition shall be reported to the Director - Division of Reactor Licensing and to the Director - Region I Compliance Office. The report shall be made prior to the expiration of the specified out-of-service period to allow the AEC to review the circumstances of the extended outage and to render a decision on extending the specified out-of-service period thereby permitting reactor operation to continue.

Basis

The Chemical and Volume Control System provides control of the Reactor Coolant System boron inventory. This is normally accomplished by using any one of the three charging pumps in series with either one of the two boric acid transfer pumps. An alternate method of boration will be to use the charging pumps taking suction directly from the refueling water storage tank.

A third method will be to depressurize and use the safety injection pumps. There are three sources of borated water available for injection through 3 different paths.

(1) The boric acid transfer pumps can deliver the boric acid tank contents to the charging pumps.

(2) The charging pumps can take suction from the refueling water storage tank. (2000 ppm boron solution. Reference is made to Technical Specification 3.3A).

(3) The safety injection pumps can take their suctions from either the refueling water storage tank or the boron injection tank.

The quantity of boric acid in storage from either the boric acid tanks or the refueling water storage tank is sufficient to borate the reactor coolant in order to reach cold shutdown at any time during core life.

Approximately 4000 gallons of the 11 1/2% to 13% by weight (20,000 ppm to 22,500 ppm of boron) of boric acid are required to meet cold shutdown conditions.

Thus, a minimum of 4400 gallons in the boric acid tanks is specified. An upper concentration limit of 13% (22,500 ppm of boron) boric acid in the tank is specified to maintain solution solubility at the specified low temperature limit of 145°F. One of two channels of heat tracing is sufficient to maintain the specified low temperature limit.

The plant operating procedures will 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. Infrequently, however, major maintenance might be required. Replacement of principal system components could necessitate outages of more than the time allowed for a system or component to be out of service. This extended maintenance specification is intended to allow a ruling by the AEC on equipment outages on a case by case basis.

Reference

FSAR - Section 9.2

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.

Specification

A. Containment Integrity

The containment integrity (as defined in Specification 1.7) shall be maintained at all times except when:

1. The reactor is in the cold shutdown condition and the shutdown margin is $\geq 1\%$ ΔK/K.

or

2. The reactor is in the cold shutdown condition and the shutdown margin is $\geq 10\%$ ΔK/K whenever the reactor vessel head is completely unbolted.

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 10% $\Delta k/k$ shutdown margin when the head is off which precludes criticality under any circumstances, even though fuel is being moved. When the reactor head is not to be removed, the specified cold shutdown margin of 1% $\Delta k/k$ precludes criticality in any occurrence.

Regarding internal pressure limitations, the containment design 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.[1] The containment can withstand an internal vacuum of 2.5 psig.[2] 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.[3]

## References


(1) FSAR - Section 14.3.5

(2) FSAR - Section 5.5

(3) FSAR - Section 5.1.1.1

## 3.8 REFUELING

### Applicability

Applies to operating limitations during refueling operations.

### Objective

To ensure that no incident could occur during refueling operations that would adversely affect public health and safety.

### Specification

A. During refueling operations the following conditions shall be satisfied:

1. The equipment door and at least one door in each personnel air lock shall be properly closed. In addition, at least one isolation valve shall be operable or locked closed in each line penetrating the containment and which provides a direct path from containment atmosphere to the outside.

2. Radiation levels in the containment and spent fuel storage areas shall be monitored continuously.

3. The core subcritical neutron flux shall be continuously monitored by the two source range neutron monitors, each with continuous visual indication in the control room and one with audible indication in the containment available whenever core geometry is being changed. When core geometry is not being changed at least one source range neutron flux monitor shall be in service.

4. At least one residual heat removal pump and heat exchanger shall be operable.

5. During reactor vessel head removal and while loading and unloading fuel from the reactor, $T_{avg}$ shall be $\leq$ 140°F and the minimum boron concentration sufficient to maintain the reactor

subcritical by at least 10% $\Delta k/k$. The required boron concentration shall be verified by chemical analysis daily.

6. Direct communication between the control room and the refueling cavity manipulator crane shall be available whenever changes in core geometry are taking place.

7. The spent fuel cask shall not be moved over spent fuel.

8. The containment vent and purge system, including the radiation monitors which initiates isolation, shall be tested and verified to be operable immediately prior to refueling operations.

9. No movement of fuel in the reactor shall be made until the reactor has been subcritical for at least ninety hours.

10. The minimum water level above the top of the core shall be at least 23 feet whenever movement of spent fuel is being made.

11. A dead-load test shall be sucessfully performed on the fuel storage refueling building crane before fuel movement begins. The load assumed by the refueling crane for this test must be equal to or greater than the maximum load to be assumed by the refueling crane during the refueling operation. A through visual inspection of the refueling crane shall be made after the dead load test and prior to fuel handling.

12. The fuel-handling building charcoal filtration system must be operating whenever spent fuel movement is being made. The fuel handling building charcoal filtration system need not be operating whenever the spent fuel has had a continuous 35 day decay period.

13. A licensed senior reactor operator shall be at the site and designated in charge of the operation whenever changes in core geometry are taking place.

B. If any of the specified limiting conditions for refueling is not met, refueling shall cease until the specified limits are met, and no operations which may increase the reactivity of the core shall be made.

Basis

The equipment and general procedures to be utilized during refueling are discussed in the FSAR. Detailed instructions, the above specified precautions, and the design of the fuel handling equipment incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety.(1) Whenever changes are not being made in core geometry, one flux monitor is sufficient. This permits maintenance of the instrumentation. Continuous monitoring of radiation levels (2 above) and neutron flux provides immediate indication of an unsafe condition. The residual heat pump is used to maintain a uniform boron concentration.

The shutdown margin indicated in Part 5 will keep the core subcritical, even if all control rods were withdrawn from the core. During refueling, the reactor refueling cavity is filled with approximately 350,000 gallons of water from the refueling water storage tank with a boron concentration of 2000 ppm. The minimum boron concentration of this water at 1564 ppm boron is sufficient to maintain the reactor subcritical by at least 10% $\Delta k/k$ in the cold condition with all rods inserted, and will also maintain the core subcritical even if no control rods were inserted into the reactor.(2) Periodic checks of refueling water boron concentration insure the proper shutdown margin. Part 6 allows the control room operator to inform the manipulator operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

In addition to the above safeguards, interlocks are utilized during refueling to ensure safe handling. An excess weight interlock is provided on the lifting hoist to prevent movement of more than one fuel assembly at a time. The spent fuel transfer mechanism can accommodate only one fuel assembly at a time.

The ninety hours decay time following plant shutdown and the 23 feet of water above the top of the core are consistent with the assumptions used in the dose calculation for the fuel handling accident. The requirement for the fuel handling building charcoal filtration system to be operating when spent fuel movement is being made provides added assurance that the offsite doses will be within acceptable limits in the event of a fuel handling accident. The additional month of spent fuel decay time will provide the same assurance that the offsite doses are within acceptable limits and therefore the charcoal filtration system would not be required to be operating.

During normal operation when the spent fuel cask is being placed in or removed from its position in the spent fuel pit, mechanical stops will be incorporated on the bridge rails which will make it impossible for the bridge of the crane to travel further north than a point directly over the spot reserved for the cask in the pit.

During normal reactor operation, the two sourthernmost spent fuel racks, each holding 25 fuel assemblies and the southern half of the rack holding 32 fuel assemblies in the southeast corner of the pit will be covered with removable stainless steel plates, to prevent the normal storage of fuel assemblies in those 66 positions closest to the south wall of the spent fuel pit. These restricted storage locations would be utilized only in the event that the total fuel assemblies are removed and 1/3 of a core from a previous refueling is present.

Thus it will be possible to handle the spent fuel cask with the 40 ton hook and to move new fuel to the new fuel elevator with the 5 ton hook, but under normal conditions it will be impossible to carry any object over the spent fuel storage area with either the 40 or 5 ton hook of the fuel storage building crane. Dead load test and visual inspection of the refueling building crane before handling spent fuel provide assurance that the crane is capable of proper operation.

The presence of a licensed senior reactor operator at the site and designated in charge provides qualified supervision of the refueling operation during changes in core geometry.

References

(1) FSAR - Section 9.5.2

(2) FSAR - Table 3.2.1-1

## 4.3 REACTOR COOLANT SYSTEM INTEGRITY TESTING

### Applicability

Applies to test requirements for Reactor Coolant System integrity.

### Objective

To specify tests for Reactor Coolant System integrity after the system is closed following normal opening, modification or repair.

### Specification

a) When the Reactor Coolant System is closed after it has been opened, the system will be leak tested at not less than 2335 psig at NDT requirements for temperature.

b) When Reactor Coolant System modifications or repairs have been made which involve new strength welds on components, the new welds will meet the requirements of ASME Section XI, IS400 and IS500.

### Basis

For normal opening the integrity of the system, in terms of strength, is unchanged. If the system does not leak at 2335 psig (Operating pressure + 100 psi: $\pm$ 100 psi is normal system pressure fluctuation), it will be leak tight during normal operation.

For repairs on components, the thorough non-destructive testing gives a very high degree of confidence in the integrity of the system, and will detect any significant defects in and near the new welds. In all cases, the leak test will assure leak tightness during normal operation.

## 4.6 EMERGENCY POWER SYSTEM PERIODIC TESTS

Applicability

Applies to periodic testing and surveillance requirements of the emergency power system.

Objective

To verify that the emergency power system will respond promptly and properly when required.

Specification

The following tests and surveillance shall be performed as stated:

A. Diesel Generators

1. Each month each diesel generator shall be manually started and synchronized to its bus or buses and shall be allowed to assume the normal bus load.

2. At each refueling outage each diesel generator shall be manually started, synchronized and loaded up to its nameplate rating.

3. At each refueling outage to assure that each diesel generator will automatically start and assume the required load within 60 seconds after the initial start signal the following shall be accomplished - by simulating a loss of all normal AC station service power supplies and simultaneously simulating a Safety Injection signal observations shall verify automatic start of each diesel generator, required bus load shedding and restoration to operation of particular vital equipment. To prevent Safety Injection flow to the core certain safeguard valves will be closed and made inoperable.

4. Each diesel generator shall be given a thorough inspection at least annually following the manufacturer's recommendations for this class of stand-by service.

The above tests will be considered satisfactory if the required minimum safeguards equipment operated as designed.

B. Diesel Fuel Tanks

A minimum oil storage of 41,000 gallons will be maintained at the station at all times.

C. Station Batteries

1. Every month the voltage of each cell, the specific gravity and temperature of a pilot cell in each battery and each battery voltage shall be measured and recorded.

2. Every 3 months each battery shall be subjected to a 24 hour equalizing charge, and the specific gravity of each cell, the temperature reading of every fifth cell, the height of electrolyte, and the amount of water added shall be measured and recorded.

3. At each time data is recorded, new data shall be compared with old to detect signs of abuse or deterioration.

4. Once a year the battery shall be subjected to a load test and a visual inspection of the plates.

## Basis

The tests specified are designed to demonstrate that the diesel generators will provide power for operation of equipment. They also assure that the emergency generator system controls and the control systems for the safeguards equipment will function automatically in the event of a loss of all normal 480v AC station service power.

The testing frequency specified will be often enough to identify and correct any mechanical or electrical deficiency before it can result in a system failure. The fuel supply is continuously monitored. An abnormal condition in these systems would be signaled without having to place the diesel generators themselves on test.

Each diesel generator has a continuous rating of 1750 kw with a 2000 hr rating of 2000 kw. Two diesels operating at their continuous rating can power the minimum safeguards loads. A minimum oil storage of 41,000 gallons will provide for operation of the minimum required engineered safeguards on emergency diesel power for a period of 168 hours.

Station batteries will deteriorate with time, but precipitous failure is extremely unlikely. The surveillance specified is that which has been demonstrated over the years to provide an indication of a cell becoming unserviceable long before it fails. The periodic equalizing charge will ensure that the ampere-hour capability of the batteries is maintained.

The annual load test for the battery together with the visual inspection of the plates will assure the continued integrity of the batteries. The batteries are of the type that can be visually inspected, and this method of assuring the continued integrity of the battery is proven standard power plant practice.

## Reference

FSAR, Section 8.2

## A-1 REACTOR COOLANT SYSTEM BORON CONCENTRATION

### Applicability

Applies to the limiting concentration of boric acid in the Reactor Coolant System and in the Safety Injection System.

### Objective

To maintain the core subcritical at all times.

### Specification

The concentration of boric acid in the Reactor Coolant System and in the refueling water storage tank and the accumulators shall not be less than 2000 ppm boron.

### Basis

With a concentration of 2000 ppm boron in the core, there will be a 4% hot shutdown margin and a 5% cold shutdown margin at all times, even if all control rods are fully withdrawn. Maintaining at least the same concentration in the accumulators and in the refueling water storage tank assures that no inadvertent action of the Safety Injection System or the Chemical and Volume Control System could cause dilution of the boron in the Reactor Coolant System.

## A.2 CONTROL ROD DRIVES

### Applicability

Applies to the status of the power supply to the control rod drive mechanisms.

### Objective

To define the status of the control rod drive power supplies.

### Specification

At all times (except during control rod testing), the power shall be disconnected from all control rod drive mechanisms.

### Basis

The reactor coolant system boron concentration required in Specification A-1 insures that complete withdrawal of all control rods will not result in criticality. Nevertheless, removal of power from the control rod drives provides even greater assurance that there can be no inadvertant approach to criticality.

## A.3 LOCKED VALVES IN THE CHEMICAL AND VOLUME CONTROL SYSTEM, SAMPLING SYSTEM AND SAFETY INJECTION SYSTEM

### Applicability

Applies to the status of various valves in the Chemical and Volume Control System, Sampling Systemand Safety Injection System.

### Objective

To insure that all potential paths of water with boron concentration below 2000 ppm are isolated from the Reactor Coolant System by locked valves.

### Specification

The following valves shall be locked closed:

| | |
|---|---|
| 311A | 383 |
| 311B | 385 |
| 323A | 386 |
| 323B | 393 |
| 340A | 394A |
| 340B | 394B |
| 343A | 315 |
| 343B | 994A |
| 282 | 1800 |
| 376 | 1841 |
| 1132 | 1802A |
| 285 | 885A & B |
| 317 | 845 |
| 318 | 1821 |
| 326 | 1831 |
| 330 | 350 |
| 339 | 841 |
| 356 | 1733 |
| 363 | PW84 |
| 365 | 873A |
| 378 | 1802B |
| 381 | |

### Basis

The valves specified isolate all lines which could conceivably serve as paths for unborated water to the Reactor Coolant System.

## A-4 AUXILIARY ELECTRICAL SYSTEMS

### Applicability

Applies to the availability of electrical power for instrumentation and for emergency boric acid addition.

### Objective

To define those conditions of electrical power availability necessary (1) to insure continuous monitoring of core subcriticality and (2) to provide for operation of emergency boric acid addition systems should then be required.

### Specification

A. At all times, there shall be:

1. 2 diesels operable with on-site supply of 19,000 gallons of fuel available on site.

2. 1 battery plus one charger and the d.c. distribution systems operable.

B. Should either of the above be out of service, the following shall be done until the components are returned to service:

1. All control rods shall be inserted.

2. All movement of fuel shall be stopped.

3. Sampling of boron concentration in the reactor coolant system shall be increased to once every 4 hours.

### Basis

A diesel would only be required in the event that normal off-site power was lost at the same time that emergency boric acid addition became necessary. One diesel is capable of providing more than adequate power to run a charging pump and auxiliary pumps which would be required for boric acid addition.

A battery is required to provide power to required plant instrumentation. Since plant instrumentation is used only to monitor the Reactor Coolant System and is not required to performaany automatic safety function, redundant batteries are not required. The actions required in the event of unavailability of a diesel or a battery provided greater assurance that there can be no inadvertent approach to criticality until the diesel or battery is returned to service.

## A-5 SURVEILLANCE REQUIREMENTS

### Operational Safety Review

#### Applicability

Applies to items directly related to safety limits and limiting conditions for fuel loading and subcritical testing.

#### Objective

To specify the minimum frequency and type of surveillance to be applied to plant equipment and conditions.

#### Specification

A. Calibration and checking of instrumentation shall be performed as specified in Table A.5-1.

B. Sampling and equipment tests shall be conducted as specified in Tables A.5-2 and A.5-3, respectively.

C. The following parameters shall be logged once a shift:

1. pressurizer level
2. reactor coolant system pressure
3. reactor coolant system temperature
4. nuclear flux source range count rate
5. all valves listed in Specification A-3 are locked
6. power to control rod drives are disconnected (except during tests)

#### Basis

#### Check

Failures such as blown instrument fuses, defective indicators, faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures are, in many cases, revealed by alarm or annunciator action, and a check supplements this type of built-in surveillance.

Based on experience in operation of both conventional and nuclear plant systems, when the plant is in operation, the minimum checking frequency of once per shift is deemed adequate for reactor instrumentation.

#### Calibration

Calibrations are performed to ensure the presentation and acquisition of accurate information.

Channels are subject only to the "drift" errors induced within the instrumentation itself and, consequently, can tolerate long intervals between calibration. Process system instrumentation errors induced by drift can be expected to remain within acceptable tolerances if recalibration is performed at intervals of six months.

Substantial calibration shifts within a channel (essentially a channel failure) will be revealed during routine checking and testing procedures.

TABLE A.5.1

Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels

| Channel Description | Check | Calibrate | Remarks |
|---|---|---|---|
| 1. Nuclear Source Range | S S | N.A. | 1. Bistable action (alarm) |
| 2. Reactor Coolant Temperature | S | S.A. | |
| 3. Pressurizer Water Level | S | S.A. | |
| 4. Pressurizer Pressure (High and Low) | S | S.A. | |
| 5. Analog Rod Position | S | S.A. | |
| 6. Rod Position Bank Counters | S | N.A. | With analog rod position |
| 7. Steam Generator Level | S | S.A. | |
| 8. Boric Acid Tank Level | W | S.A. | Bubbler tube rodded during calibration |
| 9. Refueling Water Storage Tank Level | W | S.A. | |
| 10. Containment & Fuel Storage Building Area Radiation Monitoring Systems | D | S.A. | |

TABLE A.5.1
(Con'd)

| Channel Description | Check | Calibrate | Remarks |
|---|---|---|---|
| 11. Accumulator Level | S | S.A. | |

NOTE: Specified intervals may be adjusted plus or minus 25% to accommodate normal test schedules.

S - Each Shift
D - Daily
W - Weekly
S.A. - Semi-annually
N.A. - Not applicable

TABLE A.5.2

Frequencies for Sampling Tests

| | | Check | Frequency | Maximum Time Between Tests |
|---|---|---|---|---|
| 1. | Reactor Coolant Samples | F, Cl & $O_2$ | Daily | 2 Days |
| 2. | Reactor Coolant Boron | Boron Concentration | Once a shift | 16 Hours |
| 3. | Refueling Water Storage Tank Water Sample | Boron Concentration | Once a day | 2 Days |
| 4. | Boric Acid Tank | Boron Concentration | Once a day | 2 Days |
| 5. | Accumulator | Boron Concentration | Once a day | 2 Days |

TABLE A.5.3

Frequencies for Equipment Tests

| | Check | Frequency | Maximum Time Between Tests |
|---|---|---|---|
| 1. Diesel Fuel Supply | Fuel Inventory | Weekly | 10 Days |

## A-6 LIMITING SAFETY SETTING, PROTECTIVE INSTRUMENTATION

### Applicability

Applies to the trip setting for the source range monitor.

### Objective

To provide for automatic control rod insertion in the event of an inadvertant approach to criticality.

### Specification

Whenever power is connected to the control rods the source range flux trip setting shall be less than or equal to $2x10^5$ counts/sec.

### Basis

The source range trip provides an automatic backup to the administrative controls which prevent criticality during control rod motion.

# 5 DESIGN FEATURES

## 5.1 SITE

Applicability

Applies to the location and extent of the reactor site.

Objective

To define those aspects of the site which affect the overall safety of the installation.

Specification

The minimum distance from the reactor center line to the boundary of the site exclusion area and the outer boundary of the low population zone as defined in 10 CFR 100.3 is 520 meters and 1100 meters, respectively. For the purpose of satisfying 10CFR-Part 20, the "Restricted Area" is the same as the "Exclusion Area" defined in Figure 2.2-2 of Section 2.2 of the FSAR.

## 5.2 CONTAINMENT

### Applicability

Applies to those design features of the Containment System relating to operational and public safety.

### Objective

To define the significant design features of the reactor containment structure.

### Specifications

A. Reactor Containment

1. The reactor containment completely encloses the entire reactor and reactor coolant system and ensures that an acceptable upper limit for leakage of radioactive materials to the environment is not exceeded even if gross failure of the reactor coolant system occurs. The structure provides biological shielding for both normal and accident situations.

2. The containment structure is designed for an internal pressure of 47 psig, plus the loads resulting from an earthquake producing 0.10g applied horizontally and 0.05g applied vertically at the same time.(1) The containment is also structurally designed to withstand an external pressure 2.5 psig higher than the internal pressure.

B. Penetrations

1. All penetrations through the containment reinforced concrete pressure barrier for pipe, electrical conductors, ducts and access hatches are of the double barrier type.(2)

2. The automatic Phase A containment isolation (trip) valves are actuated to the closed position either manually or by an automatically derived safety injection signal. The automatic Phase B containment isolation valves are tripped closed by automatic or manual containment spray actuation. The actuation system is designed such that no single component failure will prevent containment isolation if required.

C. Containment Systems

1. The containment vessel has an internal spray system which is capable of providing a distributed borated water spray of at least 2530 gpm. During the initial period of spray operation, sodium hydroxide would be added to the spray water to increase the removal of iodine from the containment atmosphere.(3)

2. The containment vessel has an internal air recirculation system which includes five fan-cooler units (centrifugal fans and water cooled heat exchangers), with a total heat removal capability of 106,000 Btu/sec under conditions following a loss of coolant accident.(4)

References


(1) FSAR Section 5.1

(2) FSAR Section 5.1.2.7

(3) FSAR Section 6.3

(4) FSAR Section 6.4

## 5.3 REACTOR

### Applicability

Applies to the reactor core, reactor coolant system, and emergency core cooling systems.

### Objective

To define those design features which are essential in providing for safe system operations.

### A. Reactor Core

1. The reactor core contains approximately 87 metric tons of uranium in the form of slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy-4 tubing to form fuel rods. The reactor core is made up of 193 fuel assemblies. Each fuel assembly contains 204 fuel rods.[1]

2. The average enrichment of the initial core is a nominal 2.70 weight per cent of U-235. Three fuel enrichments are used in the initial core. The highest enrichment is a nominal 3.20 weight per cent of U-235.[2]

3. Reload fuel will be similar in design to the initial core. The enrichment of reload fuel will be no more than 3.4 weight per cent of U-235.

4. Burnable poison rods are incorporated in the initial core. There are 1160 poison rods in the form of 6,12 and 16-rod clusters, which are located in vacant rod cluster control guide tubes.[3] The burnable poison rods consist of borated pyrex glass clad with stainless steel.[4]

5. There are 53 full-length RCC assemblies and 8 partial-length RCC assemblies in the reactor core. The full-length RCC assemblies contain a 142 inch length of silver-indium-cadmium alloy clad with the stainless steel. The partial-length RCC assemblies contain a 36 inch length of silver-indium-cadmium alloy with the remainder of the stainless steel sheath filled with $Al_2O_3$.[5]

B. Reactor Coolant System

1. The design of the reactor coolant system complies with the code requirements.[6]

2. All piping, components and supporting structures of the reactor coolant system are designed to Class I requirements, and have been designed to withstand the maximum potential seismic ground acceleration, 0.15g, acting in the horizontal and 0.10g acting in the vertical planes simultaneously with no loss of function.

3. The total liquid volume of the reactor coolant system, at rated operating conditions, is 11,350 cubic feet.

References

(1) FSAR Section 3.2.2
(2) FSAR Section 3.2.1
(3) FSAR Section 3.2.1
(4) FSAR Section 3.2.3
(5) FSAR Sections 3.2.1 & 3.2.3
(6) FSAR Table 4.1-9

## 5.4 FUEL STORAGE

Applicability

Applies to the capacity and storage arrays of new and spent fuel.

Objective

To define those aspects of fuel storage relating to prevention of criticality in fuel storage areas.

Specification

1. The spent fuel pit structure is designed to withstand the anticipated earthquake loadings as a Class I structure. The spent fuel pit has a stainless steel liner to insure against loss of water.

2. The new and spent fuel storage racks are designed so that it is impossible to insert assemblies in other than an array of vertical fuel assemblies with the sufficient center-to-center distance between assemblies to assure $k_{eff} \leq 0.90$ even if unborated water were used to fill the pit.

3. Whenever there is fuel in the pit (except in the initial core loading), the spent fuel storage pit is filled and borated to the concentration to match that used in the reactor cavity and refueling canal during refueling operations.

## SECTION 6

## ADMINISTRATIVE CONTROLS

### INTRODUCTION

Administrative controls relate to the organization and management procedures, record keeping, review and audit systems, and reporting that are considered necessary to provide the assurance and evidence that the plant will be managed in a dependable manner.

The administrative controls specify the administrative tools and functions necessary for the plant's safe operation. They also define the administrative action to be taken in the event operating limits or safety limits are exceeded.

### 6.1 ORGANIZATION, REVIEW AND AUDIT

A. Organization

1. Overall full-time responsibility for the safe operation of the facility shall rest with the Nuclear Plant Station Manager.

2. The Nuclear Plant Station Manager shall report to the Manager of the Nuclear Power Generation Department who, in turn, shall report to the Assistant Vice President of the Company for Power Generation Operation who is in charge of all its generating facilities.

4. The minimum functional operating organization for Unit 2 shall be complised of the following:

a During core loading operations-

1. One Senior Reactor Operator liscensed pursuant to 10 CFR55 in the containment building in charge of core loading operations.

2. One Reactor Operator liscensed pursuant to 10CFR55 in the central control room at all times.

b Subsequent to core loading operations-

1. When the reactor is in the cold shutdown condition and there is fuel in the reactor, one Reactor Operator liscensed pursuant to 10CFR55 in the central control room at all times.

2. When the reactor is in the cold shutdown condition and there is fuel in the reactor, a Senior Reactor Operator liscensed pursuant to 10CFR55 shall be present at the facility or readily available on call at all times.

3. At all other times when there is fuel in the reactor one Reactor Operator in the central control room and one Senior Reactor Operator present at the facility. Both the Reactor Operator and the Senior Reactor Operator shall be liscensed pursuant to 10CFR55.

d. A health physics technician present at the facility at all times when nuclear fuel is located therein.

5. Qualifications with regard to education and experience backgrounds and technical specialties of key supervisory personnel shall equal or surpass the minimum acceptable levels described in the "Standard for Selection and Training of Personnel for Nuclear Power Plants" Draft 9, as proposed by the American Nuclear Society.

6. The corporate organization is as shown in Figure 6.1-2.

7. Retraining and replacement training of plant personnel shall be in accordance with Section 5.5 of the "Proposed Standard for Selection and Training of Personnel for Nuclear Power Plants" ANS-3 Draft No. 9, dated 7-3-69.

B. Review and Audit

1. There shall be a Nuclear Facilities Safety Committee which shall review the operation of the facility, the operating orgainzation, the procedures for operation, changes in the facility and the conduct of tests or experiments herein.

a. Membership

The Committee shall have a membership of at least 12 persons of which a majority are independent of the Nuclear Power Generation Department and shall include technically competent persons from all departments of Consolidated Edison having a direct interest in nuclear plant design, operation or in nuclear safety. The Chairman and Vice Chairman will be Senior Officials of the Company experienced in the field of nuclear energy.

The Committee shall consist of:

The Chairman who shall be appointed by the Chairman of the Board or the President of the Company.

The Vice Chairman who shall be appointed by the Chairman of the Board or the President of the Company.

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

The following Committee Members shall be designated by the Vice President of the Company who is responsible for the functioning of the department or position stated below with the approval of the Chairman:

The Radiation Safety Officer of the Company

A medical doctor from the Medical Department having experience in nuclear medicine.

A representative from the Mechanical Engineering Department having experience in nuclear engineering with special emphasis on reactor physics.

A representative from the Nuclear Power Generation Department having experience in nuclear chemistry.

An engineer from the Fuel Department having experience with nuclear fuel.

An engineer from the Electrical Engineering Department having experience in electrical engineering related to nuclear power plants with special emphasis on instrumentation and control.

An engineer from the Mechanical Engineering Department having experience in mechanical engineering related to nuclear power plants with special emphasis on heat transfer.

A representative from the Civil Engineering Department having experience in environmental engineering.

A lawyer from the Law Department who shall be familiar with legal matters affecting nuclear power plants.

The Manager of the Nuclear Power Generation Department.

The Manager of the System Operation Department.

The Reactor Engineer at the Indian Point Station.

Outside consultants as required, appointed by the Chairman without the right to vote.

The Nuclear Plant Station Manager as a participant without the right to vote.

Each member will designate a permanent alternate to serve in his absence. The name of the alternates will be filed with the Chairman. Only the permanent member, however, will have the right to vote.

b. Minimum Meeting Frequency

The Committee shall meet not less frequently than quarterly, and at more frequent intervals at the call of the Chairman or in his absence the Vice Chairman, as required.

c. Quorum

A majority of the full committee members which shall include the Chairman or the Vice Chairman and of which a minority are from the Nuclear Power Generation Department shall constitute a quorum for meetings of the full committee.

d. Responsibilities

The Committee will:

Not less than once each year audit and report the adequacy of all procedures used in the operation, maintenance and environmental monitoring of each nuclear power plant. The audits will include on-site inspections and verifications that procedures are adhering to the Operating Licenses and Technical Specifications.

Review and report upon each emergency or infrequent condition relating to nuclear safety including as a minimum those abnormal occurrences defined in the facilities Technical Specifications.

Review and report upon the adequacy of all proposed changes in plant facilities or procedures pertaining to the operation, maintenance and environmental monitoring having safety significance, or which may constitute an "unreviewed safety question" as defined in Part 50, Title 10, Code of Federal Regulations.

Review and report upon the adequacy of nuclear safety provisions for all tests and experiments and results thereof, when such tests or experiments may constitute an "unreviewed safety question" as defined in Part 50, Title 10, Code of Federal Regulations.

Conduct not less than quarterly unannounced spot inspections of plant and montoring operations and report the results thereof.

Review and report upon any activity, the occurrence or lack of which may affect the safe operation of the nuclear plants.

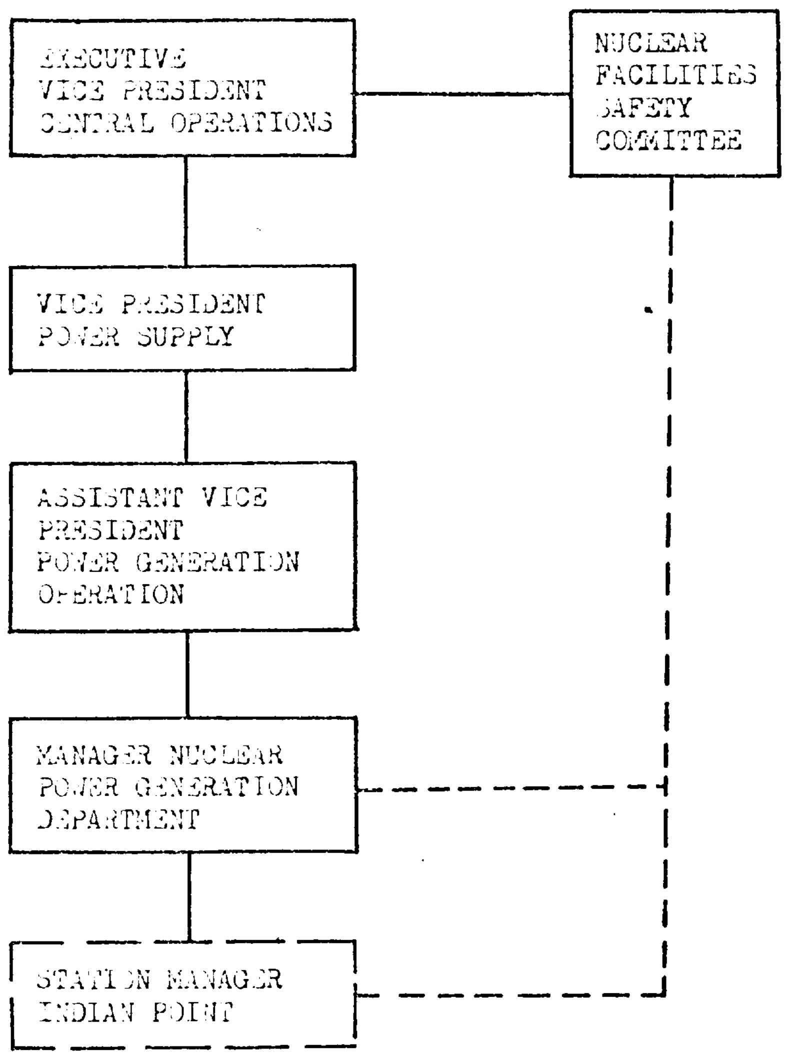
Review and report upon all proposed changes to the Technical Specifications or licenses.

At the request of the Nuclear Power Generation Manager or a Nuclear Plant Station Manager, the Committee will be promptly convened to review and act upon those nuclear safety matters deemed essential to the safe operation of the facility.

e. Authority

A Nuclear Facilities Safety Committee is constituted to advise the Executive Vice President, Central Operations, concerning the safety aspects of the operation of the nuclear power facilities. The Committee shall report to the Executive Vice President, Central Operations.

The Executive Vice President, Central Operations is responsible for the design, construction, operation and maintenance of nuclear power generation plants. The Vice President, Power Supply and, under him, the Assistant Vice President in charge of Power Generation, the Nuclear Power Generation Manager and Nuclear Plant Station Manager are responsible for the day-by-day operation and maintenance of the plant. The Nuclear Facilities Safety Committee herein established will advise the Executive Vice President and through him the President and the Chairman of the Board concerning the safety aspects of the nuclear plant operation. The Safety Committee is to be kept fully and currently informed by the Nuclear Power Generation Manager and Nuclear Plant Station Manager of all matters bearing on the safe operation of the plant.




CORPORATE ORGANIZATION

FIGURE 6.1-2

## 6.2 ACTION TO BE TAKEN IN THE EVENT OF AN ABNORMAL OCCURENCE IN PLANT OPERATION

6.2.1 Any abnormal occurence shall be promptly investigated by the Station Manager.

6.2.2 The Station Manager shall promptly notify the chairman of the Nuclear Facilities Safety Committee and the Manager of the Nuclear Power Generation Department of any abnormal occurrence.

6.2.3 The Station Manager shall prepare and submit promptly a report in writing to the Manager of the Nuclear Power Generation Department following the observation of an abnormal occurrence. Such report shall describe the circumstances leading up to, and resulting from the occurrence; and shall recommend appropriate action to prevent or reduce the probability of a repetition of occurrence. A copy of the report shall be submitted to the chairman of the Nuclear Facilities Safety Committee for review and approval of any recommendation.

6.2.4 The Vice President of Power Supply shall report the circumstances of any abnormal occurrence to the AEC as specified in Section 6.6 "Plant Reporting Requirements".

## 6.3 ACTION TO BE TAKEN IF A SAFETY LIMIT IS EXCEEDED

6.3.1 If a safety limit is exceeded, the reactor shall be shut down immediately and maintained in a safe shutdown condition until otherwise authorized by the AEC.

6.3.2 The Station Manager shall make an immediate report of the circumstances to the Manager of the Nuclear Power Generation Department and the Chairman of the Nuclear Facilities Safety Committee.

6.3.3 The Vice President of Power Supply shall report the circumstances to the AEC as specified in Section 6.6 "Plant Reporting Requirements".

6.3.4 A complete analysis of the circumstances leading up to and resulting from the situation together with recommendations to prevent a recurrence shall be prepared by the Station Manager. This report shall be submitted to the Manager of the Nuclear Power Generation Department and the Chairman of the Nuclear Facilities Safety Committee.

6.3.5 Appropriate analyses or reports shall be submitted to the AEC by the Vice President of Power supply as specified in Section 6.6 "Plant Reporting Requirements."

## 6.4 ACTIONS TO BE TAKEN PRIOR TO SPECIAL TESTS OR CHANGES

6.4.1 If the Station Manager decides to make a change in the facility or operating procedures, or to conduct a test or experiment, and concludes that the proposed change, test or experiment does not involve a change in the Technical Specifications or an unreviewed safety question, he may order the change, test or experiment to be made, shall enter a description thereof in the operating records of the facility, and shall send a copy of the instructions pertinent thereto, to the Chairman of the Nuclear Facilities Safety Committee. If the Chairman of the Committee, upon reviewing such instructions, is of the opinion that the change, test or experiment is of such a nature as to warrant consideration by the Committee, he shall order such consideration.

6.4.2 If the Station Manager desires to make a change in the facility or operating procedures or to conduct a test or experiment which in his opinion might involve a change in the Technical Specifications, might involve an unreviewed safety question or might otherwise not be in accordance with said License, he shall not order such change, test or experiment until he has referred the matter to the Nuclear Facilities Safety Committee for review and report. If the Committee is of the opinion that the proposed change, test or experiment does not require approval by the Atomic Energy Commission under the terms of said License, it shall so report in writing to the Station Manager, together with a statement of the reasons for the Committee decision and the Station Manager may then proceed with the change, test or experiment. If, on the other hand, the Committee is of the opinion that approval of the Atomic Energy Commission is required, the Committee shall prepare a request for such approval, including an appropriate safety analysis in support of the request, and forward its report and request to the Vice Presidents in charge of Engineering and Power Supply for their review with a copy to the Station Manager. One of said Vice Presidents shall thereupon forward the report and request to the Atomic Energy Commision for approval unless, after review, the Vice Presidents either (a) disagree with the opinion of the Committee that approval of the Atomic Energy Commission is required, or (b) decide that the proposed change, test or experiment is not necessary from the standpoint of Company policy or operations.

## 6.5 STATION OPERATING RECORDS

6.5.1 The following records shall be prepared and retained for five (5) years unless a longer period is required by applicable regulations. All records will be retained in a manner convenient for review.

a. Records of abnormal occurrences.

b. Records of periodic checks, inspections and/or calibration performed to verify that requirements specified under surveillance requirements are being met. All equipment failing to meet surveillance requirements and the corrective action taken will be recorded.

c. Records of reactor tests and measurements.

d. Records of changes made in the Operating Procedures.

6.5.2 The following records shall be prepared and retained for the life of the plant.

a. Records of changes made in the plant as described in the FSAR.

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

## 6.6 PLANT REPORTING REQUIREMENTS

In addition to reports required by applicable regulations, the following information shall be provided to the Atomic Energy Commission:

6.6.1 Events requiring notification within 24 hours (by telephone and telegraph to the Director of Region I Compliance Office followed by a written report within 10 days to the Director, Division of Reactor Licensing, USAEC, Washington, D.C. 20545; with a copy to the Director of Region I Compliance Office:

A. Incidents or conditions which result in exceeding a Safety Limit established in the Technical Specifications.

B. Abnormal occurrences specified in the Definitions Section of the Technical Specifications.

The written report, and to the extent possible the preliminary telephone or telegraph report, shall describe, analyze and evaluate safety implications, and outline the corrective actions and measures taken or planned to prevent recurrence of A. and B. above.

6.6.2 Events requiring reports within 30 days (in writing) to the Director, Division of Reactor Licensing, USAEC, Washington, D.C. 20545; with a copy to the Director of Region I Compliance Office:

A. Any change in transient or accident analyses, as described in the Safety Analysis Report, which involves an unreviewed safety question as defined in Paragraph 50.59 (c) of 10 CFR 50.

B. Any changes in plant operating organization which involve positions for which minimum qualifications are specified in the Technical Specifications, or in personnel assigned to these positions.

6.6.3 Events requiring reports within 60 days (in writing) to the Director, Division of Reactor Licensing, USAEC, Washington, D.C. 20545; with a copy to the Director of Region I Compliance Office:

A. Upon receipt of a new operating license or amendment to a facility license involving the planned increase in reactor power level or the installation of a new core, a summary report of plant startup and power escalation test programs and evaluations of results thereof shall be submitted within 60 days following completion of testing or commencement of commercial power, whichever comes first.

B. Any changes in the plant or corporate organizations as shown on Figures 6.1-1 and 6.1-2.

6.6.4 A Semi-Annual Station Operations Report shall be prepared and submitted to the Director, Division of Reactor Licensing, USAEC, Washington, D.C. 20545 within 60 days after the end of each reporting period.

The report shall provide the following information (summarized on a monthly basis) and shall cover the six month period or fraction thereof, ending June 30 and December 31. The due date for the first report shall be calculated from the date of initial criticality.

A. Operations Summary

1. A narrative summary of operating experience and of changes in facility design, performance characteristics (including fuel performance) and operating procedures related to safety occurring during the reporting period.

2. A summary of results of surveillance tests and inspections.

## 6.7 PLANT OPERATING PROCEDURES

6.7.1 The plant will be operated and maintained in accordance with approved procedures. Detailed written procedures with appropriate check-off lists and instructions shall be provided for the following conditions:

a. Normal startup, operation and shutdown of the complete facility and of all systems and components involving nuclear safety of the facility.

b. Refueling operations.

c. Actions taken to correct specific and foreseen potential malfunctions of systems or components, including responses to alarms, suspected primary system leaks and abnormal reactivity changes.

d. Emergency procedures involving potential or actual release of radioactivity.

e. Preventive or corrective maintenance which could affect the safety of the reactor.

f. Plant survey following an earthquake.

g. Tornado watch or tornado warning.

h. Radiation control procedures.

i. Turbine missile accident procedures.

6.7.2 Written procedures pertaining to the operations listed above shall be approved for implementation in accordance with the provisions of Section 6.1.B.

6.7.3 Temporary changes to written operating procedures, which do not change the original intent of the procedure, may be authorized by the Station Manager. Written operating procedures shall be permanently changed in accordance with the provisions of 6.4 of these specifications.

6.7.4 The Station Manager shall initiate drills at least semi-annually to ensure proficiency in emergency operations.

## 6.8 PLANT SURVEY FOLLOWING AN EARTHQUAKE

### Applicability

Applies to the inspecting of plant systems following an earthquake at the Indian Point site.

### Objective

To specify procedures that determine whether plant systems are functioning properly after the occurrence of an earthquake.

### Specification

If an earthquake shock is felt or reported to occur in the vicinity of the Indian Point Station the company's seismological consultant will be contacted within 24 hours by telephone for an evaluation of the magnitude at the Indian Point Site. The operator, after observing the earthquake, will immediately check the control boards to determine the effect, if any, on instrumentation, controls, and plant operation. An inspection will then be performed of Class II structures and equipment for visual indication of damage. An emergency evaluation will be made if damage exists or if the magnitude of the earthquake at Indian Point as reported by the seismological consultant exceeds the design basis earthquake.

### Basis

Even though the Indian Point site is a region of low seismic activity the Indian Point No. 2 plant is designed to withstand seismic activity. Appendix A of the Indian Point 2 FSAR defines this design. All equipment and structures are classified as either Class I, and Class II, or Class III.[1]

### Reference

[1] FSAR - Appendix A

A tour of inspection of Class II structures and equipment will be conducted if an earthquake is felt at Indian Point. This inspection will be coincident with a check of control and instrumentation for Class I systems in the control room. Visual signs of damage of Class II equipment and structures and indications from the control room of anomalies in the operation of Class I equipment will be the basis for further plant inspection and, if necessary, corrective action.

6.9 PLANT EMERGENCY PROGRAM
IN THE EVENT OF A TORNADO WATCH OR TORNADO WARNING

Applicability

Applies to plant tornado emergency program.

Objective

To specify plant procedures in the event of a tornado watch(1) or a tornado warning.(2)

Specification

In the event of a tornado watch the emergency foreman will notify the station. Upon notification, personnel will be assigned to listen and look for a tornado.

Upon notification of a tornado warning the gas turbine generator will be started and fuel handling operations in the fuel handling building will be halted. Should a fuel handling cask be suspended from the crane at this time, it will be set down.

Basis

Conditions that can result in a tornado can be determined by the Weather Bureau. Con Edison is in contact with local weather officials and will be advised immediately if tornado watch conditions are in effect. If a tornado is seen or if the plant operator is informed by weather officials that a tornado has occurred in the vicinity of the Indian Point site, non-essential plant operations will be halted.

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(1) Tornado watch - means meteorological conditions are favorable for the formation of a tornado.

(2) Tornado warning - means that a tornado has been sighted in the area of the plant.