50-247

APPENDIX A TO OPERATING LICENSE DPR-26
TECHINCAL SPECIFICATION AND BASES FOR
INDIAN POINT NUCLEAR GENERATING PLANT UNIT NO.2

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APPENDIX A

TO

FACILITY OPERATING LICENSE DPR-26

FOR

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT NUCLEAR GENERATING PLANT UNIT No. 2

DOCKET No. 50-247

TECHNICAL SPECIFICATIONS AND BASES

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TECHNICAL SPECIFICATIONS

1 DEFINITIONS

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

1.1 a. Rated Power

A steady state reactor thermal power of 2758 MWT.

b. Thermal Power

The total core heat transfer rate from the fuel to the coolant.

1.2 Reactor Operating Conditions

1.2.1 Cold Shutdown Condition

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

1.2.2 Hot Shutdown Condition

When the reactor is subcritical, by an amount greater than or equal to the margin as specified in Technical Specification 3.10 and T_{avg} is $\geq$ 547°F.

1.2.3 Reactor Critical

When the neutron chain reaction is self-sustaining and $k_{eff} = 1.0$.

1.2.4 Power Operation Condition

When the reactor is critical and the neutron flux power range instrumentation indicates greater than 2% of rated power.

1.2.5 Refueling Operation Condition

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

1.3 Operable

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

1.4 Protective Instrumentation Logic

1.4.1 Analog Channel

An arrangement of components and modules as required to generate a single protective action signal when required by a plant condition. An analog channel loses its identity where single action signals are combined.

1.4.2 Logic Channel

A group of relay contact matrices which operate in response to the analog channels signals to generate a protective action signal.

1.5 Degree of Redundancy

The difference between the number of operable channels and the number of channels which when tripped will cause an automatic system trip.

1.6 Instrumentation Surveillance

1.6.1 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.

1.6.2 Channel Functional Test

Injection of a simulated signal into the channel to verify that it is operable, including alarm and/or trip initiating action.

1.6.3 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 or trip, and shall be deemed to include the channel functional test.

1.7 Containment Integrity

Containment integrity is defined to exist when:

a. The required non-automatic containment isolation valves are closed and blind flanges are properly installed.

b. The equipment door is properly closed and sealed by the Weld Channel and Penetration Pressurization System.

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

d. All automatic containment isolation valves are operable or closed.

e. The containment leakage satisfies Specification 4.4.

1.8 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 uncontrolled or unplanned release of radioactive material from the site, or

d. Results in engineered safety system component failures which could render the system incapable of performing its intended safety function, or

e. Results in abnormal degradation of one of the several boundaries which are designed to contain the radioactive materials resulting from the fission process, or

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

1.9 Quadrant Power Tilt

The quadrant power tilt is defined as the ratio of maximum to average of the upper excore detector currents or the lower excore detector currents whichever is greater. If one excore detector is out of service, the three in-service units are used in computing the average.

2 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMIT, REACTOR CORE

Applicability

Applies to the limiting combinations of thermal power, Reactor Coolant System pressure and coolant temperature during four-loop and three-loop operation.

Objective

To maintain the integrity of the fuel cladding.

Specification

The combination of thermal power level, coolant pressure, and coolant temperature shall not exceed the limits shown in Figures 2.1-1 and 2.1-2 for four and three-loop operation respectively. The safety limit is exceeded if the point defined by the combination of Reactor Coolant System average temperature and power level is at any time above the appropriate pressure line.

The Region 1 fuel residence time shall be limited to 21,000 effective full power hours (EFPH) under design operating conditions. The licensee may propose to operate individual assemblies from Region 1 in excess of 21,000 EFPH by providing an analysis which includes the effect of clad flattening or a change in operating conditions. Any such analysis, if proposed, shall be approved by the Regulatory Staff prior to operation in excess of 21,000 EFPH.

Basis

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all operating conditions. This is accomplished by operating the hot region of the core within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed departure from

nucleate boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters: thermal power, reactor coolant temperature and pressure have been related to DNB through the W-3 DNB correlation. The W-3 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.30. This corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.[1]

The curves of Figure 2.1-1 and 2.1-2 represent the loci of points of thermal power, coolant system pressure and average temperature for which the DNBR is no less than 1.30. The area where clad integrity is assured is below these lines.

The curves are based on the following nuclear hot channel factors:[2]

$$F_q^N = 3.12$$

$$F_{\Delta H}^N = 1.75$$

These limiting hot channel factors are higher than those calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion.[3] The control rod insertion limits are covered by Specification 3.10. Higher hot channel factors could occur at lower power levels because additional control rods are in the core. However, the control rod insertion limits dictated by Figure 3.10-1 insure that the DNBR is always greater at partial power than at full power. For three loop operation the insertion limits of Figure 3.10-2 apply.

The hot channel factors are also sufficiently large to account for the degree of malpositioning of part-length rods that is allowed before the reactor trip set points are reduced and rod withdrawal block and load runback may be required.[2] Rod withdrawal block and load runback occurs if reactor trip setpoints are approached within a fixed limit.

The Reactor Control and Protection System is designed to prevent any anticipated combination of transient conditions that would result in a DNBR of less than 1.30.[4]

References

(1) FSAR Section 3.2.2

(2) FSAR Section 3.2.1

(3) FSAR Technical Specification 3.10

(4) FSAR Section 14.1.1

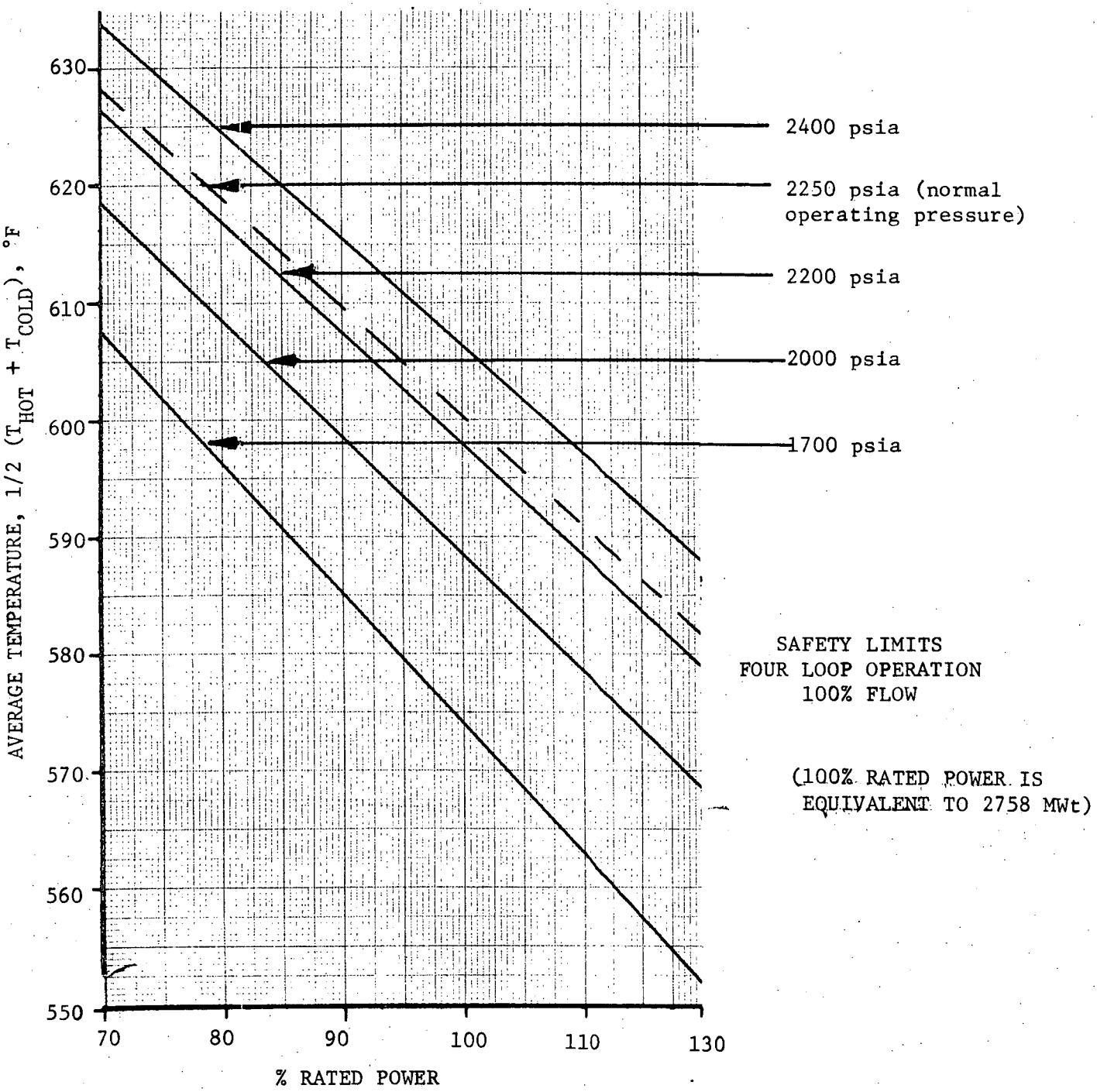
630
620
610
600
590
580
570
560
550
AVERAGE TEMPERATURE, 1/2 (T_{HOT} + T_{COLD}), °F
70
80
90
100
110
130
% RATED POWER
2400 psia
2250 psia (normal operating pressure)
2200 psia
2000 psia
1700 psia
SAFETY LIMITS
FOUR LOOP OPERATION
100% FLOW
(100% RATED POWER IS EQUIVALENT TO 2758 MWt)

FIGURE 2.1-1

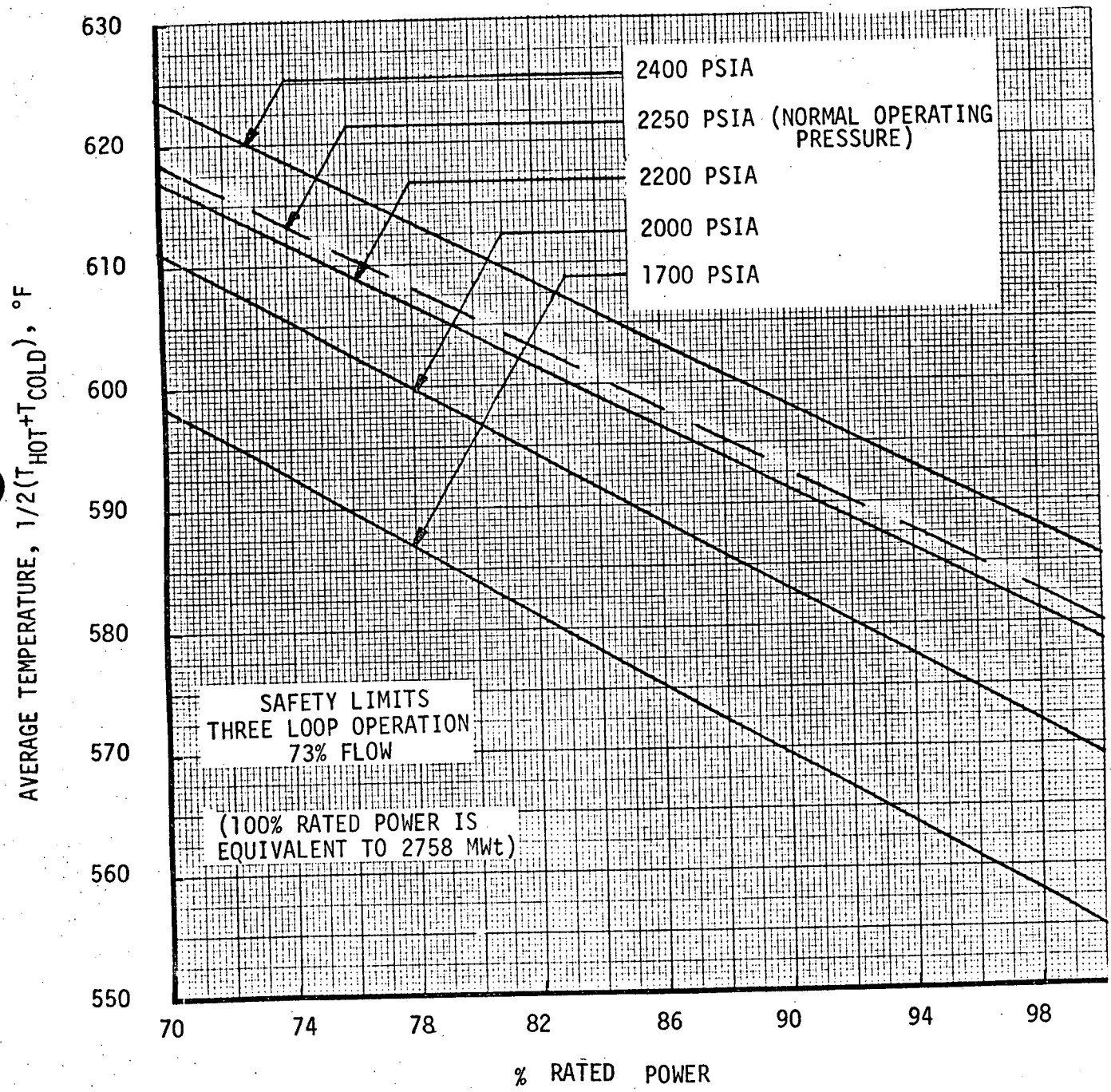
630
620
610
600
590
580
570
560
550
AVERAGE TEMPERATURE, 1/2(T_HOT+T_COLD), °F
70
74
78
82
86
90
94
98
% RATED POWER
2400 PSIA
2250 PSIA (NORMAL OPERATING PRESSURE)
2200 PSIA
2000 PSIA
1700 PSIA
SAFETY LIMITS
THREE LOOP OPERATION
73% FLOW
(100% RATED POWER IS EQUIVALENT TO 2758 MWt)

FIGURE 2.1-2

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

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Applicability

Applies to trip settings for instruments monitoring reactor power and reactor coolant pressure, temperature, flow, and pressurizer level.

Objective

To provide for automatic protective action such that the principal process variables do not exceed a safety limit.

Specification

1. Protective instrumentation for reactor trip settings shall be as follows:

   A. Startup protection

      (1) High flux, power range (low set point) - $\leq$25% of rated power.

   B. Core limit protection

      (1) High flux, power range (high set point) - $\leq$109% of rated power.

      (2) High pressurizer pressure - $\leq$2385 psig.

(3) Low pressurizer pressure - $\geq$1700 psig.

(4) Overtemperature ΔT

$$\leq \Delta T_o \; [[K_1 - K_2 \; (T - T') + K_3 \; (P - P') - f(\Delta I)]$$

where

ΔT_o = Indicated ΔT at rated power

T = Average temperature, °F

T' = Indicated T_{avg} at nominal condition at rated power, 570°F

P = Pressurizer pressure, psig

P' = 2235 psig

$K_1 \leq 1.17$	Four Loop Operation	$K_1 \leq 1.13$	Three Loop Operation
$K_2 \geq 0.012$		$K_2 \geq 0.0117$	
$K_3 \leq 0.00056$		$K_3 \leq 0.00048$	

and $f(\Delta I)$ as defined in (5) below.

(5) Overpower ΔT

$$\leq \Delta T_o \; [K_4 - K_5 \; \frac{dT}{dt} - K_6 \; (T - T') - f \; (\Delta I)]$$

where

ΔT_o = Indicated ΔT at rated power

T = Average temperature, °F

T' = Indicated T_{avg} at nominal condition at rated power, 570°F

K_4 $\leq$ 1.19

K_5 = Zero for decreasing average temperature

K_5 $\geq$ 0.188, for increasing average temperature (sec/°F)

K_6 $\geq$ 0.0019 for $T \geq T'$; K_6 = 0 for $T < T'$

$\frac{dT}{dt}$ = Rate of change of T_{avg}

and $f(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

1. For $(q_t - q_b)$ within the range between ΔI_1 and ΔI_2 given in the table below, $f(\Delta I) = 0$ (where q_t and q_b are percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of rated power).

2. For each percent that $(q_t - q_b)$ is less than ΔI_1, the Delta-T trip setpoint shall be automatically reduced by 4.5% of its value at rated power. For each percent that $(q_t - q_b)$ is greater than ΔI_2, the Delta-T trip setpoint shall be automatically reduced by 2% of its value at rated power.

ΔI_1 and ΔI_2 are linear functions of the gain K_4. The proper limits on ΔI_1 and ΔI_2 shall be obtained from the following table which gives the allowable values corresponding to the actual value of K_4.

K_4	ΔI_1	ΔI_2
$\leq$ 1.01	$\geq$ -16.0	$\leq$ +16
1.04	$\geq$ -15.33	$\leq$ +14.5
1.07	$\geq$ -14.66	$\leq$ +13
1.10	$\geq$ -14.0	$\leq$ +11.5
1.13	$\geq$ -13.33	$\leq$ +10
1.16	$\geq$ -12.66	$\leq$ +8.5
1.19	$\geq$ -12	$\leq$ +7

(6) Low reactor coolant loop flow:

(a) $\geq$90% of normal indicated loop flow

(b) Low reactor coolant pump frequency - $\geq$ 57.5 cps

(7) Undervoltage - $\geq$ 70% of normal voltage

C. Other reactor trips

(1) High pressurizer water level - $\leq$ 92% of span

(2) Low-low steam generator water level - $\geq$ 5% of narrow range instrument span.

2. Protective instrumentation settings for reactor trip interlocks shall satisfy the following conditions:

A. The reactor trips on low pressurizer pressure, high pressurizer level, and low reactor coolant flow for two or more loops shall be unblocked when:

1) Power range nuclear flux $\geq$ 10% of rated power, or

2) Turbine first stage pressure $\geq$ 10% of equivalent full load.

B. The single loop loss of flow reactor trip may be bypassed when the power range nuclear instrumentation indicates $\leq$ 60% of rated power. The single loop loss of flow reactor trip may be bypassed below 75% of rated power only when the overtemperature ΔT trip setpoint has been adjusted to the three-loop operation value given in 2.3.1.B-4 above. The resetting of the overtemperature ΔT trip shall be performed by the Technical Service Bureau under the direct supervision of the Operations Staff of Consolidated Edison Company.

Basis

The high flux reactor trips provide redundant protection in the power range for a power excursion beginning from low power. This trip was used in the safety analysis.(1)

The power range nuclear flux reactor trip high set point protects the reactor core against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The prescribed set point, with allowance for errors, is consistent with the trip point assumed in the accident analysis.(2) (3)

The source and intermediate range reactor trips do not appear in the specification as these settings are not used in the transient and accident analysis (FSAR Section 14). Both trips provide protection during reactor startup. The former is set at about 10^{+5} counts/sec and the latter at a current proportional to approximately 25% of rated full power.

The high and low pressure reactor trips limit the pressure range in which reactor operation is permitted. The high pressurizer pressure reactor trip is backed up by the pressurizer code safety valves for overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The low pressurizer pressure reactor trip also trips the reactor in the unlikely event of a loss of coolant accident. Its setting limit is consistent with the value assumed in the loss of coolant analysis.[4]

The overtemperature Delta-T reactor trip provides core protection against DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided only that (1) the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds)[5], and (2) pressure is within the range between the high and low pressure reactor trips. With normal axial power distribution, the reactor trip limit, with allowance for errors[2], is always below the core safety limit as shown on Figure 2.1-1. If axial peaks are greater than design, as indicated by difference between top and bottom power range nuclear detectors, the reactor trip limit is automatically reduced.[6][7]

In order to operate with a reactor coolant loop out of service (three-loop operation), only the overtemperature ΔT trip set-point calculation would have to be modified. Sustained operation with a reactor coolant loop out of service is a rare event. When this mode of operation is chosen, the variables K_1, K_2 and K_3 must be adjusted and the overtemperature ΔT trip channels must be recalibrated. These adjustments and calibrations must be made in the protection system racks and are performed as is done for four-loop operation before initial startup and during normal calibration procedures. The set-point adjustments are made based on limits for reduced power three-loop operation and provide sufficient margin for three-loop operation.

The overpower Delta-T reactor trip prevents power density anywhere in the core from exceeding 112% of design power density, as described in Section 7.2.3 and 14.1.2 and includes corrections for axial power distribution, change in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. The specified set points meet this requirement and include allowance for instrument errors.[2]

The low flow reactor trip protects the core against DNB in the event of a loss of one or two reactor coolant pumps. The undervoltage reactor trip protects the core against DNB in the event of a loss of two or more reactor coolant pumps. The set points specified are consistent with the values used in the accident analysis.[8] The low frequency reactor coolant pump trip also protects against a decrease in flow. The specified set point assures a reactor trip signal by opening the reactor coolant pump breaker before the low flow trip point is reached.

The high pressurizer water level reactor trip protects the pressurizer safety valves against water relief. Approximately 1600 ft^3 of water (39.75 ft above the lower instrument tap) corresponds to 92% of span. The specified set point allows margin for instrument error and transient level overshoot beyond their trip setting so that the trip function prevents the water level from reaching the safety valves.

The low-low steam generator water level reactor trip protects against postulated loss of feedwater accidents. The specified set point assures that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays for the Auxiliary Feedwater System.[9]

Specified reactor trips are blocked at low power where they are not required for protection and would otherwise interfere with normal plant operations. The prescribed set point at which these trips are unblocked assures their availability in the power range where needed.

Above 10% power, an automatic reactor trip will occur if two reactor coolant pumps are lost during operation. Above 60% power, an automatic reactor trip will occur if any pump is lost. This latter trip will prevent the minimum value of the DNB ratio, DNBR, from going below 1.30 during normal operational transients and anticipated transients when only three loops are in operation and the overtemperature ΔT trip setpoint is adjusted to the value specified for four loop operation. When the overtemperature ΔT trip setpoint is adjusted to the value specified for three loop operation, the trip at 75 percent power

will prevent the minimum value of the DNBR from going below 1.30 during normal operational transients and anticipated transients when only three loops are in operation.

The turbine and steam-feedwater flow mismatch trips do not appear in the specification as these settings are not used in the transient and accident analysis (FSAR Section 14).

References

(1) FSAR 14.1.1
(2) FSAR 14.1.2
(3) FSAR Table 7.4.2
(4) FSAR 14.3.1
(5) FSAR 14.1.2
(6) FSAR 7.2
(7) FSAR 3.2.1
(8) FSAR 14.1.6
(9) FSAR 14.1.9

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 $\pm$ 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.

Reference

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 (1.61×10^6 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 2.4×10^{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 2.4×10^{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 1.61×10^6 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 1.2×10^{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

C. MINIMUM CONDITIONS FOR CRITICALITY

1. Except during low power physics tests, the reactor shall not be made critical at any temperature above which the moderator temperature coefficient is positive.

2. In no case shall the reactor be made critical below DTT +10°F, where the value of DTT+10°F is as determined in part B of this specification.

3. When the reactor coolant temperature is below the minimum temperature specified in 1. above, the reactor shall be subcritical by an amount equal to or greater than the potential reactivity insertion due to depressurization.

4. The reactor shall be maintained subcritical by at least 1% until normal water level is established in the pressurizer.

Basis:

During the early part of the initial fuel cycle, the moderator temperature coefficient is calculated to be slightly positive at coolant temperatures below the power operating range.[(1)(2)] The moderator coefficient at low temperatures will be most positive at the beginning of life of the fuel cycle, when the boron concentration in the coolant is the greatest. Later in the life of the fuel cycle, the boron concentrations in the coolant will be lower and the moderator coefficients will be either less positive or will be negative. At all times, the moderator coefficient is negative in the power operating range.[(1)(2)] Suitable physics measurements of moderator coefficients of reactivity will be made as part of the startup program to verify analytic predictions.

The requirement that the reactor is not to be made critical when the moderator coefficient is positive has been imposed to prevent any unexpected power excursion during normal operations as a result of either an increase of moderator temperature or decrease of coolant pressure. This requirement is waived during low power physics tests to permit measurement of reactor moderator coefficient and other physics design parameters of interest. During physics tests, special operating precautions will be taken.

The requirement that the reactor is not to be made critical below DTT + 10°F provides increased assurance that the proper relationship between reactor coolant pressure and temperature will be maintained during system heatup and pressurization whenever the reactor vessel is in the nil-ductility temperature range. Heatup to this temperature will be accomplished by operating the reactor coolant pumps.

If the shutdown margin specified in 3.1.C-3 is maintained, there is no possibility of an accidental criticality as a result of a decrease of coolant pressure.

The requirement for bubble formation in the pressurizer when the reactor has passed the threshold of 1% subcriticality will assure that the Reactor Coolant System will not be solid when criticality is achieved.

References:

1. FSAR Table 3.2.1-1
2. FSAR Figure 3.2.1-9

D. MAXIMUM REACTOR COOLANT ACTIVITY

Specification

1. The total specific activity of the reactor coolant excluding tritium due to nuclides with half-lives of more than 30 minutes, shall not exceed $60/\bar{E}$ μCi/cc, whenever the reactor is critical or the average reactor coolant temperature is greater than 500°F. ($\bar{E}$ is the weighted average of the beta and gamma energies per disintegration in Mev.)

Basis

The specified limit provides protection to the public against the potential release of reactor coolant activity to the atmosphere, as demonstrated by the following analysis of a steam generator tube rupture accident.

Rupture of a steam generator tube would allow a portion of the reactor coolant activity to enter the secondary system. The major portion of this activity is noble gases which are diverted to the containment within a few seconds after the air ejector monitors high activity signal. The activity release to atmosphere is not significant.

In the event the air ejector discharge is not diverted to the containment a portion of the reactor coolant noble gas activity would be released to the atmosphere through the secondary system. Activity could continue to be released until the operator would reduce the primary system pressure below the lowest setpoint of the secondary relief valves and could isolate the faulty steam generator. The worst credible set of circumstances is considered to be a double-ended break of a single tube, with the air ejector discharging to the atmosphere, followed by isolation of the faulty steam generator by the operator within 30 minutes after the event. During that time approximately one-eighth of the total reactor coolant could be released to the Steam and Feedwater System.[1]

The limiting off-site dose is the whole-body dose resulting from immersion in the cloud containing the released activity. Radiation would include both gamma and beta radiation. The gamma dose is dependent on the finite size and configuration of the cloud. However, the analysis will employ the simple model of a semi-infinite cloud, which gives an upper limit to the potential gamma dose. The semi-infinite cloud model is applicable to the beta dose, because of the short range of beta radiation in air. The effectiveness of clothing as shielding against beta radiation is neglected and therefore the analysis model also gives an upper limit to the potential beta dose.

The combined gamma and beta dose from a semi-infinite cloud is given by:

$$\text{Dose (rem)} = 1/2 \{\bar{E} \cdot A \cdot V \cdot X/Q\ (3.7 \times 10^{10}) \cdot (1.33 \times 10^{-11})\}$$

Where: $\bar{E}$ = weighted average energy of betas and gammas per disintegration (Mev/dis)

A = primary coolant activity (Ci/m^3)

V = primary coolant volume released to the secondary side ($44.5\ m^3$).

$X/Q = 7.5 \times 10^{-4}\ sec/m^3$, the 0-2 hr. dispersion coefficient at the site boundary[2]

3.7×10^{10} dis/sec - Ci

$1.33 \times 10^{-11}\ rem/Mev/m^3$

The resulting dose is 0.5 rem at the site boundary when A is equal to $\frac{60}{\bar{E}}$, which is the expression used in this specification.

If the air ejector discharge is diverted to the containment, the only activity released to atmosphere is that contained in the steam flow to the turbine gland seal ($5000\ \frac{lb}{hr}$). For this case the activity release to atmosphere during the 30 minute period would be 1.1% of the values given above. It

is concluded that a tube rupture accident would not result in significant radiation exposure.

The basis for the 500°F temperature contained in the specification is that saturation pressure corresponding to 500°F, 680.8 psia, is well below the pressure at which the atmospheric relief valves on the secondary side would be actuated.

Calculations required to determine $\bar{E}$ will consist of the following:

1. Quantitative measurement in units of μCi/cc of radionuclides with half lives longer than 30 minutes making up at least 95% of the total activity in the primary coolant.

2. A determination of the beta and gamma decay energy per disintegration of each nuclide determined in (1) above by applying known decay energies and schemes. (Table of Isotopes, Sixth Edition, March 1968).

3. A calculation of $\bar{E}$ by appropriate weighting of each nuclides beta and gamma energy with its concentration as determined in (1) above.

References

(1) FSAR Table 9.2-5
(2) FSAR Section 11.1.3
(3) FSAR Table 14.2.4
(4) FSAR Table 2.7.3

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

F. LEAKAGE OF REACTOR COOLANT

Specification

1. If leakage of reactor coolant is indicated by the means available such as water inventory balance, monitoring equipment or direct observation, a follow up evaluation of the safety implications shall be initiated as soon as practicable but no later than within 4 hours. Any indicated leak shall be considered to be a real leak until it is determined that either (1) a safety problem does not exist or (2) that the indicated leak cannot be substantiated by direct observation or other indication.

2. If the indicated leakage is substantiated and is not evaluated as safe or is determined to exceed 10 GPM, reactor shutdown shall be initiated as soon as practicable but no later than within 24 hours after the leak was first detected.

3. The nature of the leak as well as the magnitude of the leak shall be considered in the safety evaluation. If plant shutdown is necessary per specification 2 above, the rate of shutdown and the conditions of shutdown shall be determined by the safety evaluation for each case and justified in writing as soon thereafter as practicable. The safety evaluation shall assure that the exposure to offsite personnel to radiation from the primary system coolant activity is within the guidelines of 10 CFR 20.

4. The reactor shall not be restarted until the leak is repaired or until the problem is otherwise corrected.

5. When the reactor is critical and above 2% power, two reactor coolant leak detection systems of different principles shall be in operation, with one of the two systems sensitive to radioactivity. The system sensitive to radioactivity may be out-of-service for 48 hours provided two other systems are available.

Basis:

Water inventory balances, monitoring equipment, radioactive tracing, boric acid crystalline deposits, and physical inspections can disclose reactor coolant leaks. Any leak of radioactive fluid, whether from the reactor coolant system primary boundary or not can be a serious problem with respect to in-plant radioactivity contamination and cleanup or it could develop into a still more serious problem; and therefore, first indications of such leakage will be followed up as soon as practicable.

Although some leak rates on the order of GPM may be tolerable from a dose point of view, especially if they are to closed systems, it must be recognized that leaks in the order of drops per minute through any of the walls of the primary system could be indicative of materials failure such as by stress corrosion cracking. If depressurization, isolation and/or other safety measures are not taken promptly, these small leaks could develop into much larger leaks, possibly into a gross pipe rupture. Therefore, the nature of the leak, as well as the magnitude of the leakage must be considered in the safety evaluation.

When the source of leakage has been identified, the situation can be evaluated to determine if operation can safely continue. This evaluation will be performed by the Plant Operating Staff and will be documented in writing and approved by either the General Superintendent or his designated alternate. Under these conditions, an allowable primary system leakage rate of 10 gpm has been established. This explained leakage rate of 10 gpm is also well within the capacity of one charging pump and makeup would be available even under the loss of off-site power condition.

If leakage is to the containment, it may be identified by one or more of the following methods:

a. The containment air particulate monitor is sensitive to low leak rates. The rates of reactor coolant leakage to which the instrument

is sensitive are 0.025 gpm to greater than 10 gpm, assuming corrosion product activity and no fuel cladding leakage. Under these conditions, an increase in coolant leakage of 1 gpm is detectable within 1 minute after it occurs.

b. The containment radiogas monitor is less sensitive and is used as a backup to the air particulate monitor. The sensitivity range of the instrument is 10^{-3} μc/cc to 10^{-6} μc/cc. Assuming a constant background radioactivity in the containment atmosphere due to normal leakage of reactor coolant with equilibrium fission product gaseous activity, a 1 gpm coolant leak would double the background in about two hours time.

c. The humidity detectors. This method provides a backup to a. and b. This instrument will be sensitive to incremental increases of water leakage to the containment atmosphere on the order of 0.25 gpm per F degree of dewpoint temperature increase.

d. A leakage detection system which determines leakage losses from all water and steam systems within the containment. This system collects and measures moisture condensed from the containment atmosphere by cooling coils of the main air recirculation units. This system provides a dependable and accurate means of measuring integrated total leakage, including leaks from the cooling coils themselves which are part of the containment boundary. Condensate flows from approximately 1 gpm to 30 gpm per detector can be measured by this system.

Leaks less than 1 gpm may be determined by periodic observation of the water accumulation in the standpipes of the condensate collection system.

As described above, the four reactor coolant leak detection systems are based on 3 different principles, i.e. activity, humidity and condensate flow measurements. Two systems of different principles provide, therefore, diversified ways of detecting leakage to the containment.

If leakage is to be another closed system, it will be detected by the plant radiation monitors and/or inventory control.

References

FSAR Sections 11.2.3 and 14.2.4

REACTOR COOLANT SYSTEM
HEATUP LIMITATIONS

Applicable to 1.61×10^6 MWt - Days

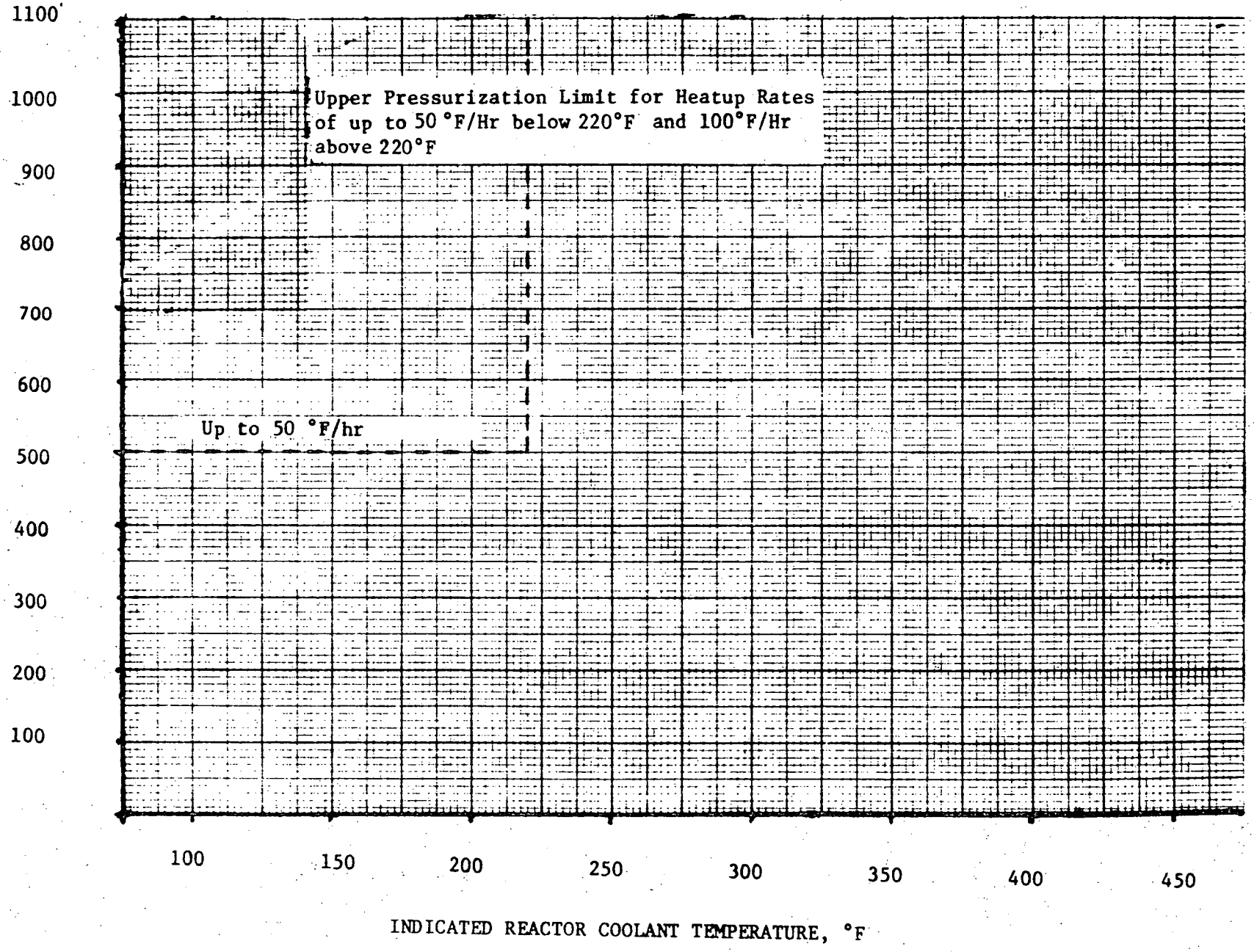

FIGURE 3.1-1

REACTOR COOLANT SYSTEM
COOLDOWN LIMITATIONS

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

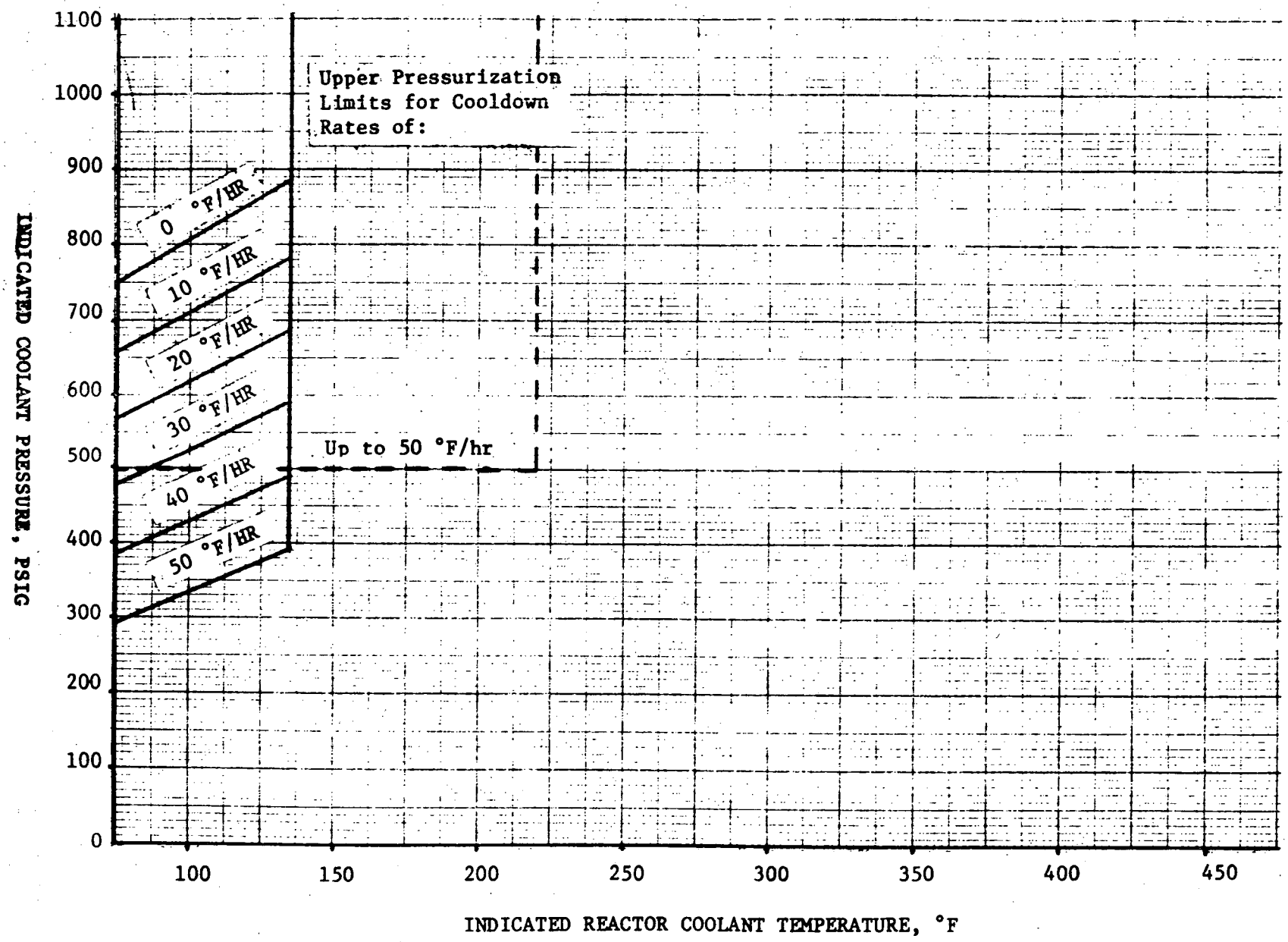

FIGURE 3.1-2

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.

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.

Reference

FSAR - Section 9.2

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.

Specification

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 tank contains not less than 350,000 gal. of water with a boron concentration of at least 2000 ppm.

   b. The boron injection tank contains not less than 1000 gal. of a 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. Two channels of heat tracing, shall be available for the flow path. Valves 1821 and 1831 shall be open and valves 1822A and 1822B shall be closed, except during short period of time when they can be cycled to demonstrate their operability.

c. The four accumulators are pressurized to at least 600 psig and each contains a minimum of 700 ft^3 and a maximum of 715 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 together with their associated piping and valves are operable.

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

g. Two residual heat exchangers together with their associated piping and valves are operable.

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 blocked closed by de-energizing the valve motor operators.

i. The four accumulator isolation valves shall be blocked open by de-energizing the valve motor operators.

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.

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 demonstrated to be 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 demonstrated to be operable.

c. One residual heat removal 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 demonstrated to be operable.

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 gal. of solution with a sodium hydroxide concentration of not less than 30% 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. Fan cooler unit 23, 24, or 25 or the flow path for fan cooler unit 23, 24, or 25 may be out of service during normal reactor operation for a period not to exceed 24 hours, provided both containment spray pumps are demonstrated to be operable.

OR

Fan cooler unit 21 or 22, or the flow path for fan cooler unit 21 or 22 may be out of service during normal reactor operation for a period not to exceed 7 days provided both containment spray pumps are demonstrated daily to be operable.

b. One containment spray pump may be out of service during normal reactor operation, for a period not to exceed 24 hours, provided the five fan cooler units are operable and the remaining containment spray pump is demonstrated to be operable.

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

C. Isolation Valve Seal Water System

The isolation valve seal water system shall be operable when the reactor is critical.

D. Weld Channel and Penetration Pressurization System

The weld channel and penetration pressurization system shall be operable when the reactor is critical.

E. Component Cooling System

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

   a. Two component cooling pumps on busses supplied by different diesels 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 two operable component cooling pumps may be out of service provided the pump is restored to operable status within 24 hours.

   b. 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 demonstrated to be operable.

   c. 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. Service Water System

1. The reactor shall not be made critical unless the following condition is met:

   Three service water pumps on the designated essential header together with their associated piping and valves are operable.

2. If during power operation one of the three service water pumps on the designated essential header or any of their associated piping or valves is found inoperable, the operator shall immediately proceed to place in service an essential service water system which meets the requirements of 3.3.F-1. If an essential service water system can not be restored within eight hours, the reactor shall be placed in cold shutdown condition.

G. Hydrogen Recombiner System

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 unit train located outside of the containment which may be inoperable, provided it is under repair and can be made operable if needed.

   b) The containment atmosphere sampling system including the sampling pump, piping and valves are 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 4.5.I.C.1.

2. During power operation, the requirement 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. If the requirements are not satisfied within an additional 48 hours, the reactor shall be placed in the cold 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 seven days, provided the other recombiner unit is demonstrated to be operable.

   b) One containment atmosphere sampling line may be inoperable for a period not to exceed seven days, provided the other sampling lines are demonstrated to be operable.

   c) The containment atmosphere sampling pump may be inoperable for a period not to exceed seven days, provided a spare pump is available at the site for service if required.

H. 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.H-1 may be modified to allow one cable tunnel ventilation fan to be inoperable for seven days, provided the other fan is daily demonstrated to be 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 safeguard 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 Specifications 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. For a single component to be inoperable 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 (a) the inoperable component is not repaired within the specified allowable time period, or (b) 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) Assuring with high reliability that the safeguard system will function properly if required to do so.

2) Allowances 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.

The line from the Boron Injection Tank to the high head pump suction piping is provided with four motorized valves; two valves in series with each other and two valves in parallel with each other. Valves 1821 and 1831 are in series and are redundant to each other to assure tank isolation after boron injection, i.e., at least one valve must close. Valves 1822 A and B are in parallel and are redundant to each other, to assure an open path for boron injection following a safety injection signal.

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 to occur 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.[3] 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.[2] 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 6.2-6 of the FSAR.

The containment cooling and iodine removal functions are provided by two independent systems: (a) fan-coolers plus charcoal filters and (b) 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 85°F).[4] 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 off-site 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 off-site 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 off-site 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 off-site 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 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]

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]

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 analyses.[9]

Two full rated 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 of the systems 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 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 95°F. The second fan will start if 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 below a 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 daily testing of 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.

References

(1) FSAR Section 9
(2) FSAR Section 6.2
(3) FSAR Section 6.2
(4) FSAR Section 6.3
(5) FSAR Section 14.3.5
(6) FSAR Section 1.2
(7) FSAR Section 8.2
(8) FSAR Section 9.6.1
(9) FSAR Section 14.3

3.4 STEAM and POWER CONVERSION SYSTEM

Applicability

Applies to the operating status of the Steam and Power Conversion System.

Objective

To define conditions of the turbine cycle steam-relieving capacity. Auxiliary Feedwater System and Service Water System operation is necessary to ensure the capability to remove decay heat from the core.

Specification

A. The reactor shall not be heated above 350°F unless the following conditions are met:

(1) A minimum ASME code approved steam-relieving capability of twenty (20) main steam valves shall be operable (except for testing).

(2) Two of the three auxiliary feedwater pumps must be operable.

(3) A minimum of 360,000 gallons of water in the condensate storage tanks and a backup supply from the city water supply.

(4) System piping and valves directly associated with the above components operable.

(5) The main steam stop valves are operable and capable of closing in five seconds or less.

(6) The total iodine activity of I-131 and I-133 on the secondary side of the steam generator shall be less than or equal to 0.15 μCi/cc.

B. If during power operations any of the conditions of 3.4.A above can not be met within 48 hours the operator shall start to shutdown and cool the reactor below 350°F using normal operating procedures.

Basis

A reactor shutdown from power requires removal of core decay heat. Immediate decay heat removal requirements are normally satisfied by the steam bypass to the condensers. Thereafter, core decay heat can be continuously dissipated via the steam bypass to the condenser as feedwater in the steam generator is converted to steam by heat absorption. Normally, the capability to feed the steam generators is provided by operation of the turbine cycle feedwater system.

The twenty main steam safety valves have a total combined rated capability of 15,108,000 lbs/hr. The total full power steam flow is 13,283,000 lbs/hr, therefore twenty (20) main steam safety valves will be able to relieve the total steam flow if necessary.

In the unlikely event of complete loss of electrical power to the station, decay heat removal would continue to be assured by the availability of either the steam-driven auxiliary feedwater pump or one of the two motor-driven auxiliary steam generator feedwater pumps, and steam discharge to the atmosphere via the main steam safety valves and atmospheric relief valves. One motor-driven auxiliary feedwater pump can supply sufficient feedwater for removal of decay heat from the plant. The minimum amount of water in the condensate storage tank is the amount needed for 24 hours at hot standby. When the condensate storage supply is exhausted, city water will be used.

The limit on secondary coolant total iodine activity of I-131 and I-133 is based on a postulated release of secondary coolant equivalent to the contents of four steam generators to the atmosphere due to a net load rejection with loss-of-offsite power. The limiting dose for this case would result from radioactive iodine in the secondary coolant. I-131 and I-133 are the dominant isotopes because of their low MPC's in air and because the other shorter-lived isotopes cannot build up to significant concentrations in the secondary coolant under the limits of primary system leak rate and activity. One tenth of the iodine in the secondary

coolant is assumed to reach the site boundary making allowance for plate-out and retention in water droplets. The inhalation dose at the site boundary is then as follows:

$$\text{Dose(rem)} = \frac{C \cdot V}{10} \cdot B(t) \cdot \chi/Q \cdot DCF$$

where: C = secondary coolant activity (0.15 μCi/cc = 0.15 Ci/m^3)

V = water volume in four steam generators ($7416\ ft^3 = 210\ m^3$)

B(t) = breathing rate ($3.47 \times 10^{-4}\ m^3/sec$)

χ/Q = $7.5 \times 10^{-4}\ sec/m^3$

DCF = 1.00×10^6 rem/Ci Iodine (131 and 133) inhaled

The resultant dose is less than 1.0 rem.

Reference

FSAR - Section 10.4 and 14.1.9

3.5 INSTRUMENTATION SYSTEMS

Operational Safety Instrumentation

Applicability:

Applies to plant instrumentation systems.

Objectives:

To provide for automatic initiation of the Engineered Safety Features in the event that principal process variable limits are exceeded, and to delineate the conditions of the plant instrumentation and safety circuits necessary to ensure reactor safety.

Specification:

3.5.1 The Engineered Safety Features initiation instrumentation setting limits shall be as stated in Table 3-1.

3.5.2 For on-line testing or instrumentation channel failure, plant operation at rated power shall be permitted to continue in accordance with Tables 3-2 through 3-4. No more than one channel of a particular protection channel set shall be tested at the same time. By definition, an instrumentation channel failure shall not be regarded as a channel being tested.

3.5.3 In the event the number of channels of a particular function in service falls below the limits given in the column entitled Minimum Operable Channels, or Minimum Degree of Redundancy cannot be achieved, operation shall be limited according to the requirement shown in Column 5 of Tables 3-2 through 3-4.

3.5.4 In the event of sub-system instrumentation channel failure permitted by specification 3.5.2, Table 3-2 through 3-4 need not be observed during the short period of time the operable sub-system channels are tested where the failed channel must be blocked to prevent unnecessary reactor trip.

3.5.5 The cover plate on the rear of the safeguard panel, in the control room shall not be removed without the authorization from the operations staff. If a cover is removed, the event must be reported in the Semi-Annual Station Operation Report in accordance with Specification 6.6.4.D.

Basis

Instrumentation has been provided to sense accident conditions and to initiate operation of the Engineered Safety Features[1].

Safety Injection System Actuation

Protection against a Loss of Coolant or Steam Break accident is brought about by automatic actuation of the Safety Injection System which provides emergency cooling and reduction of reactivity.

The Loss of Coolant Accident is characterized by depressurization of the Reactor Coolant System and rapid loss of reactor coolant to the containment. The Engineered Safety Features have been designed to sense the effects of the Loss of Coolant accident by detecting low pressurizer pressure and level and generates signals actuating the SIS active phase based upon the coincidence of these signals. The SIS active phase is also actuated by a high containment pressure signal (Hi-Level) brought about by loss of high enthalpy coolant to the containment. This actuation signal acts as a backup to the low pressurizer pressure and level signal actuation of the SIS and also adds diversity to protection against loss of coolant.

Signals are also provided to actuate the SIS upon sensing the effects of a steam line break accident. Therefore, SIS actuation following a steam line break is designed to occur upon sensing high differential steam pressure between any two steam generators or upon sensing high steam line flow in coincidence with low reactor coolant average temperature of low steam line pressure.

The increase in the extraction of RCS heat following a steam line break results in reactor coolant temperature and pressure reduction. For this reason protection against a steam line break accident is also provided by coincident low pressurizer pressure and level signals actuating safety injection.

Protection is also provided for a steam line break in the containment by actuation of SIS upon sensing high containment pressure.

SIS actuation injects highly borated fluid into the Reactor Coolant System in order to counter the reactivity insertion brought about by cooldown of the reactor coolant which occurs during a steam line break accident.

Containment Spray

The Engineered Safety Features actuation system also initiate containment spray upon sensing a high containment pressure signal (Hi-Hi Level). The containment spray acts to reduce containment pressure in the event of a loss of coolant or steam line break accident inside the containment. The spray cools the containment directly and limits the release of fission products by absorbing iodine should it be released to the containment.

Containment spray is designed to be actuated at a higher containment pressure (approximately 50% of design containment pressure) than the SIS (2.0 psig). Since spurious actuation of containment spray is to be avoided, it is automatically initiated only on coincidence of Hi-Hi Level containment pressure sensed by both sets of two-out-of-three containment pressure signals and the derived S. I. signal provided for its actuation.

Steam Line Isolation

Steam line isolation signals are initiated by the Engineered Safety Features closing all steam line stop valves. In the event of a steam line break, this action prevents continuous, uncontrolled steam release from more than

one steam generator by isolating the steam lines on high containment pressure (Hi-Hi Level) or high steam line flow. Protection is afforded for breaks inside or outside the containment even when it is assumed that there is a single failure in the steam line isolation system.

Feedwater Line Isolation

The feedwater lines are isolated upon actuation of the Safety Injection System in order to prevent excessive cooldown of the reactor coolant system. This mitigates the effect of an accident such as steam break which in itself causes excessive coolant temperature cooldown.

Feedwater line isolation also reduces the consequences of a steam line break inside the containment, by stopping the entry of feedwater.

Setting Limits

1. The Hi-Level containment pressure limit is set at 2.0 psig containment pressure. Initiation of Safety Injection protects against loss of coolant[2] or steam line break[3] accidents as discussed in the safety analysis.

2. The Hi-Hi Level containment pressure limit is set at about 50% of design containment pressure. Initiation of Containment Spray and Steam Line Isolation protects against large loss of coolant[2] or steam line break accidents[3] as discussed in the safety analysis.

3. The pressurizer low pressure limit is set substantially below system operating pressure limits. However, it is sufficiently high to protect against a loss of coolant accident as shown in the safety analysis.[2]

4. The steam line high differential pressure limit is set well below those differential pressure expected in the event of a large steam line break accident as shown in the safety analysis.[3]

5. The high steam line flow limit is set approximately 20% of the full steam flow at no load and at 120% of full steam flow at full load,

with the steam flow differential pressure measurement linearly programmed between no load and full load in order to protect against large steam line break accidents. The coincident low Tavg setting limit for SIS and steam line isolation initiation is set below its hot shutdown value. The coincident steam line pressure setting limit is set below the full load operating pressure. The safety analysis shows that these settings provide protection in the event of a large steam line break.[3]

Instrument Operating Conditions

During plant operations, the complete instrumentation systems will normally be in service. Reactor safety is provided by the Reactor Protection System, which automatically initiates appropriate action to prevent exceeding established limits. Safety is not compromised, however, by continuing operation with certain instrumentation channels out of service since provisions were made for this in the plant design. This specification outlines limiting conditions for operation necessary to preserve the effectiveness of the Reactor Control and Protection System when any one or more of the channels is out of service.

Almost all reactor protection channels are supplied with sufficient redundancy to provide the capability for channel calibration and test at power. Exceptions are backup channels such as reactor coolant pump breakers. The removal of one trip channel on process control equipment is accomplished by placing that channel bistable in a tripped mode; e.g., a two-out-of-three circuit becomes a one-out-of-two circuit. The nuclear instrumentation system channels are not intentionally placed in a tripped mode since the test signal is superimposed on the normal detector signal to test at power. Testing of the NIS power range channel requires: (a) bypassing the Dropped Rod protection from NIS, for the channel being tested; and (b) defeating the ΔT protection CHANNEL SET that is being fed from the NIS channel and (c) defeating the power mismatch section of T_{avg} control channels when the appropriate NIS channel is being tested. However, the Rod Position System

and remaining NIS channels still provide the dropped-rod protection. Testing does not trip the system unless a trip condition exists in a concurrent channel.

Reference

(1) FSAR - Section 7.5
(2) FSAR - Section 14.3
(3) FSAR - Section 14.2.5

TABLE 3-1

ENGINEERED SAFETY FEATURES INITIATION INSTRUMENT SETTING LIMITS

No.	FUNCTIONAL UNIT	CHANNEL	SETTING LIMIT
1.	High Containment Pressure (Hi Level)	Safety Injection	$\leq$ 2.0 psig
2.	High Containment Pressure (Hi-Hi Level)	a. Containment Spray b. Steam Line Isolation	$\leq$ 30 psig
3.	Pressurizer Low Pressure and Low Level	Safety Injection	$\geq$ 1700 psig $\geq$ 5 per cent instrument span
4.	High Differential Pressure Between Steam Lines	Safety Injection	$\leq$ 150 psi
5.	High Steam Flow in 2/4	a. Safety Injection	$\leq$ 20% @ (Of full steam flow at zero load)
	Steam Lines Coincident with Low T_{avg} or Low Steam Line Pressure	b. Steam Line Isolation	$\leq$ 120% @ (Of full steam flow at full load) $\geq$ 540°F T_{avg} $\geq$ 600 psig steam line pressure

TABLE 3-2

REACTOR TRIP INSTRUMENTATION LIMITING OPERATING CONDITIONS

NO.	FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MIN. OPERABLE CHANNELS	4 MIN. DEGREE OF REDUN-DANCY	5 OPERATOR ACTION IF CONDITIONS OF COLUMN 3 or 4 CANNOT BE MET
1.	Manual	2	1	1	0	Maintain hot shutdown
2.	Nuclear Flux Power Range	4	2	3	2	Maintain hot shutdown
2.a	Nuclear Flux Power Range	4	2	2	1	For zero power physics tests only
3.	Nuclear Flux Intermediate Range	2	1	1*	0	Maintain hot shutdown
4.	Nuclear Flux Source Range	2	1	1**	0	Maintain hot shutdown
5.	Overtemperature ΔT	4	2	3	2	Maintain hot shutdown
6.	Overpower ΔT	4	2	3	2	Maintain hot shutdown
7.	Low Pressurizer Pressure	4	2	3	2	Maintain hot shutdown
8.	Hi Pressurizer Pressure	3	2	2	1	Maintain hot shutdown
9.	Pressurizer-Hi Water Level	3	2	2	1	Maintain hot shutdown

TABLE 3-2 (Continued)

		1	2	3	4	5
10.	Low Flow Loop ≥75% F.P.	3/loop	2/loop (any loop)	2/operable loop	1/operable loop	Maintain hot shutdown
	Low Flow Two Loops 10-75% F.P.	3/loop	2/loop (any two loops)	2/operable loop	1/operable loop	
11.	Lo Lo Steam Generator Water Level	3/loop	2/loop	2/loop	1/loop	Maintain hot shutdown
12.	Undervoltage 6.9 KV Bus	1/bus	2	3	2	Maintain hot shutdown
13.	Low frequency 6.9 KV Bus	1/bus	2	3	2	Maintain hot shutdown ***
14.	Quadrant power tilt monitors	2	NA	1	0	Log individual upper and lower ion chamber currents once/shift and after load change >10%
15.	Turbine trip (overspeed protection)	3	2	2	1	Maintain hot shutdown

TABLE 3-2 (Continued)

* If two of four power channels greater than 10% F.P., channels are not required.

** If one of two intermediate range channels greater than 10^{-10} amps, channels are not required.

*** 2/4 trips all four reactor coolant pumps.

F.P. = Rated Power

TABLE 3-3

INSTRUMENTATION OPERATING CONDITION FOR ENGINEERED SAFETY FEATURES

NO.	FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MIN. OPERABLE CHANNELS	4 MIN. DEGREE OF REDUN-DANCY	5 OPERATOR ACTION IF CONDITIONS OF COLUMN 3 or 4 CANNOT BE MET
1	SAFETY INJECTION					
a.	Manual	2	1	1	0	Cold Shutdown
b.	High Containment Pressure (Hi Level)	3	2	2	1	Cold Shutdown
c.	High Differential Pressure Between steam Lines	3/steam line	2/steam line	2/steam line	1/steam line	Cold Shutdown
d.	Pressurizer Low Pressure and Low Level*	3**	1**	2**	1	Cold Shutdown
e.	High Steam Flow in 2/4 Steam Lines Coincident with Low T_{avg} or Low Steam Line Pressure	2/line	2/line	1/line in each of 3 lines	2	Cold Shutdowm
		4 T_{avg} Signals	2	3	2	
		4 Pressure Signals	2	3	2	
2	CONTAINMENT SPRAY					
a.	Manual	2	2	2	0***	Cold Shutdown
b.	High Containment Pressure (Hi Hi Level)	2 sets of 3	2 of 3 in each set	2 per set	1/set	Cold Shutdown

* Permissible bypass if reactor coolant pressure less than 2000 psig.
** Each channel has two separate signals.
*** Must actuate 2 switches simultaneously.

TABLE 3-4

INSTRUMENT OPERATING CONDITIONS FOR ISOLATION FUNCTIONS

NO.	FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MIN. OPERABLE CHANNELS	4 MIN. DEGREE OF REDUN-ANCY	5 OPERATOR ACTION IF CONDITIONS OF COLUMN 3 or 4 CANNOT BE MET
1.	CONTAINMENT ISOLATION					
a.	Automatic Safety Injection (Phase A)	See Item No. 1 of Table 3-3				Cold Shutdown
b.	Containment Pressure (Phase B)	See Item No. 2 of Table 3-3				Cold Shutdown
c.	Manual					
	Phase A one out of two	2	1	1	0	Cold Shutdown
	Phase B	See Item 2a of Table 3-3				Cold Shutdown
2.	STEAM LINE ISOLATION					
a.	High Steam Flow in 2/4 Steam Lines Coincident with Low T_{avg} or Low Steam Line Pressure	See Item No. 1(e) of Table 3-3				Cold Shutdown
b.	High Containment Pressure (Hi Hi Level)	See Item No. 2b of Table 3-3				Cold Shutdown
c.	Manual	1/loop	1/loop	1/loop	0	Cold Shutdown
3.	FEEDWATER LINE ISOLATION					
a.	Safety Injection	See Item No. 1 of Table 3-3				

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% $\Delta K/K$.

or

2. The reactor is in the cold shutdown condition and the shutdown margin is $\geq$10% $\Delta K/K$ 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

3.7 AUXILIARY ELECTRICAL SYSTEMS

Applicability

Applies to the availability of electrical power for the operation of plant auxiliaries.

Objective

To define those conditions of electrical power availability necessary (1) to provide for safe reactor operation and (2) to provide for the continuing availability of engineered safety features.

Specification

A. The reactor shall not be made critical without:

1. Two 138 kv lines to Buchanan fully operational.

2. The 6.9 kv buses 5 and 6 energized from the 138 kv source.

3. One 13.8 kv source fully operational and the 13.8/6.9 kv transformer available to supply 6.9 kv power.

4. The four 480-volt buses 2A, 3A, 5A and 6A energized and the bus tie breakers between buses 5A and 2A and between buses 3A and 6A open.

5. Three diesel generators operable with on-site supply of 19,000 gallons of fuel available in the individual storage tanks and 22,000 gallons of fuel available on-site other than the normal supply tanks.

6. Both batteries plus two chargers and the d.c. distribution systems operable.

B. During power operation, the following components may be inoperable:

1. Power operation may continue for seven days if one diesel is inoperable provided the 138 kv and the 13.8 kv sources of off-site power are available and the remaining diesel generators are tested daily to ensure operability and the engineered safety features associated with these diesel generator buses are operable.

2. Power operation may continue for 24 hours, if the 138 kv or the 13.8 kv source of power is lost, provided the three diesel generators are operable. This operation may be extended beyond 24 hours provided the failure is reported to the AEC within the subsequent 24-hour period with an outline of the plans for restoration of off-site power.

3. One battery may be inoperable for 24 hours provided the other battery and two battery chargers remain operable with one battery charger carrying the dc load of the failed battery's supply system.

C. The requirements of Specification 3.7.A may be modified for an emergency "Black Start" of the unit by using the requirements of either Specification 3.7.C.1 or 3.7.C.2 below:

1. a. All 138 kv lines to Buchanan de-energized.
   b. The 13.8 kv line de-energized.
   c. The 6.9 kv buses 5 and 6 energized from the on-site gas turbine through the 13.8/6.9 kv transformer.
   d. The four 480-volt buses 2A, 3A, 5A and 6A energized from the diesels and the tie breakers between buses 5A and 2A and between buses 3A and 6A open.
   e. Three diesel generators operable with on-site supply of 19,000 gallons of fuel available in the individual storage tanks and 22,000 gallons of fuel available on-site other than the normal supply tanks and supplying 480-volt buses.
   f. Both batteries plus two chargers and the d.c. distribution systems operable.
   g. The 480-volt tie breakers 52/2A, 52/3A, 52/5A and 52/6A open.

2. a. Establish 138 kv bus sections at Buchanan with at least 37 MW power (nameplate rating) from any combination of gas turbines at Buchanan and on-site.

   b. Two 138 kv lines to Buchanan energized from the gas turbines with breakers to Millwood and Orange and Rockland open.

   c. The 13.8 kv line to Buchanan operable and the 13.8/6.9 kv transformer available to supply 6.9 kv power.

   d. The 6.9 kv buses energized from the 138 kv source.

   e. The four 480-volt buses 2A, 3A, 5A and 6A energized and the bus tie breakers between buses 5A and 2A and between buses 3A and 6A open.

   f. Three diesel generators operable with on-site supply of 19,000 gallons of fuel available in the individual storage tanks and 22,000 gallons of fuel available on-site other than the normal supply tanks.

   g. Both batteries plus two chargers and the d.c. distribution system operable.

Basis

The electrical system equipment is arranged so that no single contingency can inactivate enough safeguards equipment to jeopardize the plant safety. The 480-volt equipment is arranged in four buses. The 6900-volt equipment is supplied from six buses.

In addition to the unit transformer, three separate sources supply station service power to the plant.(1)

The plant auxiliary equipment is arranged electrically so that multiple items receive their power from different sources. The charging pumps are supplied from the 480-volt buses Nos 3A, 5A, and 6A. The five containment fans are divided among the 480-volt buses. The two residual heat pumps are on separate 480-volt buses. Valves are supplied from separate motor control centers.

The station auxiliary transformer or the gas turbine is capable of providing sufficient power for plant startup. The station auxiliary transformer can supply the required plant auxiliary power during normal operation.

The bus arrangements specified for operation ensure that power is available to an adequate number of safeguards auxiliaries. With additional switching, more equipment could be out of service without infringing on safety.

Two diesel generators have sufficient capacity to start and run at design load the minimum required engineered safeguards equipment.(1) The minimum diesel fuel oil inventory at all times is maintained to assure the operation of two diesels carrying the load of the minimum required engineered safeguards equipment for at least eighty hours.(2) Additional fuel oil suitable for use in the diesel generators will be stored on site. The minimum storage of 22,000 gallons will assure operation of two diesels for ninety hours at the minimum load for engineered safeguards. Commercial oil supplies and trucking facilities exist to assure deliveries within one day's notice. One battery charger shall be in service on each battery so that the batteries will always be at full charge in anticipation of a loss-of-ac power incident. This insures that adequate d.c. power will be available for starting the emergency generators and other emergency uses.

The plant can be safely shutdown without the use of off-site power since all vital loads (safety systems, instruments, etc.) can be supplied from the emergency diesel generators.

Any two of three diesel generators, the station auxiliary transformer or the separate 13.8 to 6.9 kv transformer are each capable of supplying the minimum safeguards loads and therefore provide separate sources of power immediately available for operation of these loads. Thus, the power supply system meets the single failure criteria required of the safety systems.

Conditions of a system-wide blackout could result in a unit trip. Since normal off-site power supplies as required in Specification 3.7.A are not available for startup, it is desirable to be able to blackstart this unit with on-site power supplies as a first step in restoring the system to an operable status and restoring power to customers for essential service. Specification 3.7.C.1 provides for startup using the on-site gas turbine to supply the 6.9 kv loads and the diesels to supply the 480-volt loads. Tie breakers between the 6.9 kv and 480-volt systems are open so that the diesels would not be jeopardized in the event of any incident and would be able to to continue to supply 480-volt safeguards power. The scheme consists

of starting two reactor coolant pumps, one condensate pump, 2 circulating water pumps and necessary auxiliaries to bring the unit up to approximately 10% power. At this point, loads can be assumed by the main generator and power supplied to the system in an orderly and routine manner.

This Specification (3.7.C.2) is identical with normal startup requirements as specified in 3.7.A except that off-site power is supplied exclusively from gas turbines with a minimum total power of 37 MW (nameplate rating) which is sufficient to carry out normal plant startup.

Reference

(1) FSAR - Section 8.2.1

(2) FSAR - Section 8.2.3

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 1615 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) "Fuel Densification - Indian Point Nuclear Generating Station Unit No. 2," dated January 1973, Table 3.3.

3.9 EFFLUENT RELEASE

Applicability

Applies to the release of radioactive liquids and gases from the plant.

Objective

To define the conditions for release of radioactive wastes to the circulating water discharge and to the plant vent to assure that any radioactive material released is kept as low as practicable and, in any event within the limits of 10CFR20.

Specification

A. General

1. It is expected that releases of radioactive material in effluents will be kept at small fractions of the limits specified in 20.106 of 10CFR20. At the same time the licensee is permitted the flexibility of operation, compatible with considerations of health and safety, to assure that the Public is provided a dependable source of power even under unusual operating conditions which may temporarily result in releases higher than such small fractions, but still within limits specified in 20.106 of 10CFR20. It is expected that in using this operational flexibility under unusual operating conditions the licensee will exert his best efforts to keep levels of radioactive material in effluents as low as practicable.

2. Plant equipment shall be used in conjuction with developed operating procedures to maintain surveillance of radioactive gaseous and liquid effluents produced during normal reactor operations and expected operational occurences in an effort to maintain radioactive releases to unrestricted areas as low as practicable.

3. A report shall be submitted to the Commission at the end of each six-months' period of operation as required under Specification

6.6.4. If quantities of radioactive material released during the reporting period are unusual for normal reactor operations, including expected operational occurences, the report shall cover this specifically. On the basis of such reports and any additional information the Commission may obtain from the licensee or others, the Commission may from time to time require the licensee to take such action as the Commission deems appropriate.

B. Liquid Effluents

1. The maximum instantaneous release rate of radioactive liquid effluents from the site shall be such that the concentration of radionuclides in the circulating water discharge does not exceed the limits specified in 10CFR20, Appendix B, for unrestricted areas.

2. Prior to release of radioactive effluents, a sample shall be taken, and analyzed to provide the data necessary to assure compliance with B.(1) above.

3. During release of radioactive liquid effluents, at least one condenser circulating water pump shall be in operation.

4. During release of radioactive liquid effluents, the gross activity liquid discharge monitor shall be in operation, except that the monitor may be out-of-service for 48 hours, provided that a sample shall be taken during release of each batch of discharge line effluent and analyzed.

C. Gaseous Effluents

1. The maximum instantaneous release rate of gaseous effluents for the site shall be limited as follows:

$$\left(\frac{\chi}{Q}\right)_1 \sum_i \frac{Q_{1i}}{(MPC)_i} + \left(\frac{\chi}{Q}\right)_2 \sum_i \frac{Q_{2i}}{(MPC)_i} \leq 1.0$$

where:

i refers to any radioisotope

Q_{1i} is the release rate (Ci/sec) of any radioisotope i from Unit No. 1

Q_{2i} is the release rate (Ci/sec) of any radioisotope i from Unit No. 2

$(MPC)_i$ in units of μCi/cc as listed in Column 1, Table II of Appendix B 10CFR20, except that for isotopes of iodine and particulates with half lives greater than 8 days, the values of $(MPC)_i$ shall be reduced by a factor of 700.

$(\frac{X}{Q})_1$ and $(\frac{X}{Q})_2$ are the meterological dispersion coefficients (Sec/m^3) for Units No. 1 and No. 2 respectively at the site releasing the effluent from the plant vent, air ejector discharge, and blowdown tank vent when applicable.

$(\frac{X}{Q})_1 = 5.88 \times 10^{-7}$ sec/m^3

$(\frac{X}{Q})_2 = 2.5 \times 10^{-5}$ sec/m^3

2. Prior to release of gaseous effluents, the contents of the gas holdup tank shall be sampled and analyzed to provide the necessary data to assure compliance with Specification 3.9.C.1 above.

3. During release of gaseous effluent to the plant vent, the following conditions shall be met:

   a. At least one auxiliary building exhaust fan shall be in operation.

   b. The plant vent monitor shall be in operation and the vent halogen particulate monitor shall be in operation except that the plant vent monitor may be out-of-service for 48 hours. Should the vent monitor fail immediate action to stop gas decay tank release will be made.

4. The inventory of noble gases in any gas tank shall not exceed 16,500 curies of equivalent Xe-133.

5. Gaseous waste in the gas decay tank shall have as a mininum 20 days of decay time except for low radioactivity gaseous waste resulting from purge and fill operations associated with refueling and reactor startup.

6. During power operation the air ejector discharge monitor may be inoperable for 48 hours. When the monitor is inoperable samples shall be taken from the air ejector discharge and analyzed for gross activity on a daily basis, except when there is indication of primary to secondary leakage the sample shall be taken and analyzed for gross activity once per shift.

7. During the first indication of primary to secondary leakage, a determination of the partition factor for the blowdown tank shall be made. Whenever there is indication of primary to secondary leakage and any steam generator is being blown down, the blowdown line monitor shall be operable, except that it may be inoperable for 48 hours provided samples shall be taken once per shift of the blowdown effluent and analyzed for gross activity.

Basis

Liquid wastes from the radioactive Waste Disposal System are diluted in the Circulating Water System discharge prior to release to the river.(1) With all six pumps operating, the rated capacity of the Circulating Water System is 840,000 gpm. Loss os one circulating water pump reduces the nominal flow rate by about 20%. The actual circulating water flow under various operating conditions will be calculated from the head differential across the pumps and the manufacturer's head-capacity curves. The concentrations in the circulating water discharge will be calculated from the measured concentration in the waste condensate tank, the flow rate of the waste condensate pumps, and the flow in the Circulating Water System.

It is expected that the Plant Operating Procedures will allow releases of radioactive material and effluents to be small fractions of the limits specified in 10CFR20 and it is expected that the actual liquid release rates will result in a concentration in the circulating water discharge of less than 1/10 MPC. Thus, discharge of liquid wastes at the specified concentrations will not result in significant exposure to members of the Public as a result of consumption of drinking water from the river, even if the effects of potable water treatment systems on reducing radioactive concentration of the water supply is neglected.

Buildup of long-lived radioisotopes in the river and reconcentration by aquatic organisms in the human food chain has also been considered. Using conservatively high estimates of reconcentration of radioisotopes in fish and of human consumption of fish, it is concluded that the release of liquid wastes may equal the 10CFR20 guidelines without causing any identificable problems. While some species of rooted vegetation, and filter feeding molluscs, concentrate some of the radioactive components of a reactor effluent in the Hudson, none of these species are used for human or animal consumption. Fish, on the other hand, while possible sources of food, do not demonstrate accumulation of the nuclides in question. For both maganese and cobalt there is a natural barrier to absorption in the gut of fish which restricts their uptake of these elements. In fact, much of the reported concentration of the radio elements may be located only in undigested gut residues rather than in the fish flesh which may be consumed. Hence, the potential contamination of diet from this source is miniscule.[4] This will be continually monitored by the environmental surveillance program (as defined in Specification 4.10). However, because of the flow in the Hudson River[2], it is not anticipated that any appreciable reconcentration will occur.

Prior to release to the atmosphere, gaseous wastes from the radioactive Waste Disposal System are mixed in the plant vent with the flow from at least one of two auxiliary building exhaust fans. Further dilution then occurs in the atmosphere.

The formula prescribed in Specification 3.9.C.1 takes into account combined releases from the site, and assures that at any point on or beyond the site boundary the requirements of 10CFR20 will be satisfied. Atmospheric dilution

is taken into account with the χ/Q's for Indian Point Units No. 1 and No. 2 being based on the worst combination of sector yearly average meteorology and sector distance to the site boundary. For Indian Point Unit No. 1 alone, the value of χ/Q of 5.88×10^{-7} Sec/m^3 would result in just achieving 10CFR20 limits at the site boundary. For Indian Point Unit No. 2 alone, the value of χ/Q of 2.5×10^{-5} sec/m^3 would result in just achieving 10CFR20 limits at the site boundary. The combined formula in Specification 3.9.C.1, however, would require the release rates for any radioisotope, Q_{1i} and Q_{2i}, to be limited for consideration of joint releases being limited to 10CFR20 from the site.

Restricting the maximum inventory of noble gases in any gas or liquid tank to 16,500 curies equivalent Xe-133 (or 15% of the total maximum Reactor Coolant System inventory), will result in a total off-site exposure of less than 0.5 rem for complete release of the noble gas activity stored in the tank.(3)

References

(1) FSAR Section 10.2.4

(2) FSAR Section 2.5

(3) FSAR Section 14.2.3

(4) Development of a biological monitoring system and pesticide residues in the lower Hudson River. - M. Eisenbud and G. P. Howells - Institute of Environmental Medicine New York University Medical Center - October 10, 1969.

3.10 CONTROL ROD AND POWER DISTRUBUTION LIMITS

Applicability:

Applies to the operation of the control rods and power distribution limits.

Objective:

To ensure (1) core subcriticality after a reactor trip, (2) limited potential reactivity insertions from a hypothetical control rod ejection, and (3) an acceptable core power distribution during power operation.

Specification:

3.10.1 Control Rod Insertion Limits

3.10.1.1 When the reactor is subcritical prior to startup, the hot shutdown margin shall be at least that shown in Figure 3.10-3. The shutdown margin as used here is defined as the amount by which the reactor core would be subcritical at hot shutdown conditions if all control rods were tripped, assuming that the highest worth control rod remained fully withdrawn, and assuming no changes in xenon, boron, or part-length rod position.

3.10.1.2 When the reactor is critical, except for physics tests and control rod exercises indicated in Table 4.1-3, the shutdown control rods shall be fully withdrawn.

3.10.1.3 When the reactor is critical, except for physics tests and control rod exercises, the control group rods shall be no further inserted than the limits shown by the lines on Figure 3.10-1 for 4 loop operation on Figure 3.10-2 for 3 loop operation.

3.10.1.4 During physics tests and control rod exercises indicated in Table 4.1-3, the insertion limits need not be observed, but the Figure 3.10-3 must be observed except for rod worth measurements.

3.10.1.5 The part-length rods shall not be more than 70% inserted.

3.10.2 Power-Distribution Limits and Misaligned Control Rod

3.10.2.1 The moveable detector system shall be used to confirm power distribution, such that design limits are not exceeded, after initial fuel loading and after each fuel reloading, prior to operation of the plant above 75% of rated power.

If the core is operating above 75% power with one excore nuclear channel out of service, then the core quadrant power balance shall be determined once a day by at least one of the following means:

a. Moveable detectors (at least 2 thimbles per quadrant)
b. Core-exit thermocouples (at least 4 thermocouples per quadrant)

In addition, when operating above 50% power, the moveable detector system shall be used to confirm power distribution monthly.

3.10.2.2 At all times, except for physics tests at 90% of rated power or less, the hot channel factors must meet the following limits:

$F_Q^N \leq 2.62\ [1 + 0.2\ (1-P)]$ in the indicated flux difference range of +7 to -12 percent.

$F_{\Delta H}^N \leq 1.65\ [1 + 0.2\ (1-P)]$

where P is the fraction of full power at which the core is operating.

The measured values, with due allowance for measurement error must be corrected by including a penalty as shown on Figure 3.10-4 (at the approximate core location) to account for fuel densification effects before comparison with the limiting values above.

If the hot channel factors exceed these limits, the reactor power and high neutron flux trip setpoints shall be reduced by 1 percent for every percent excess over F_Q^N or $F_{\Delta H}^N$, whichever is limiting. If the hot channel factors cannot be corrected within one day, the

overpower ΔT and overtemperature ΔT trip setpoints shall be similarly reduced.

3.10.2.3 If, except for physics tests at 90% of rated power or less, the quadrant to average power tilt exceeds a value of T%, where T is 5% or can be increased according to:

$$\frac{F_{xy}}{1.435} \left[1 + 2 \left(\frac{T}{100} - 0.02\right) \right] = \frac{1.06}{P}$$

where F_{xy} is either 1.435 or the appropriate value of the unrodded horizontal plane peaking factor measured by a moveable in-core detector flux map and T may not exceed 10%, or if a part-length or full-length control rod is more than 15 inches out of alignment with its bank, then within 2 hours:

a. the situation shall be corrected, or

b. the hot channel factors will be determined and the power level and trips adjusted according to Specification 3.10.2.2; or

c. if the hot channel factors are not determined within 2 hours, the power shall be reduced from 100% power, 2% for each percent of quadrant tilt in excess of T%.

3.10.2.4 If the quadrant to average power tilt exceeds 10%, except for physics tests, the power level and high neutron flux trip setpoint will be reduced from 100% power, 2% for each percent of quadrant tilt.

3.10.2.5 The cause for any quadrant power tilt above T% which persists for more than 24 hours or which recurs intermittently, and which exceeds T% shall be determined. If the cause of the tilt cannot be determined within 5 days of operation, the reactor power level shall be restricted so as not to exceed 50% of rated power.

If the cause of the tilt is determined, continued operation at full power or at a power level determined by 3.10.2.2 above, shall be permitted.

3.10.2.6 If after a period of 24 hours, the power tilt ratio in 3.10.2.2 is not corrected to less than T%:

a. An evaluation of the cause of the discrepancy shall be made and reported to the Atomic Energy Commission, and

b. The nuclear overpower, overpower ΔT and overtemperature ΔT trips shall be reduced one percent for each percent the operating power level has been reduced.

c. If the design hot channel factors for rated power are exceeded and the power is greater than 10%, the Atomic Energy Commission shall be notified and the nuclear overpower, overpower ΔT and overtemperature ΔT trips shall be reduced one percent for each percent the hot channel factors exceeds the rated power design values.

3.10.2.7 If the quadrant to average power tilt ratio exceeds 1.15, the reactor shall be put in the hot shutdown condition utilizing normal operating procedures and the Atomic Energy Commission notified. Low power operation for purposes of accomplishing correction of tilt is permitted (less than 50% of rated power).

3.10.2.8 Except during physics tests, the following power distribution restrictions must be maintained:

a. At rated power, the indicated axial flux difference must be maintained with +7 percent and -12 percent.

b. If, at rated power, the indicated axial flux difference exceeds the permissible range defined above for a period of more than eight hours, the situation shall be corrected or the reactor power shall be reduced 2 percent, for each

percent the flux difference exceeds the permissible positive range, and reduced 4.5% for each percent in the negative range.

c. For every 2 percent below full power, the permissible flux difference range is extended by 1 percent in the positive range, and 0.44% in the negative range.

3.10.3 Rod Drop Time

3.10.3.1 The drop time of each control rod shall be no greater than 1.8 seconds at full flow and operating temperature from the beginning of rod motion to dashpot entry.

3.10.4 Inoperable Control Rods

3.10.4.1 A control rod shall be deemed inoperable if (a) the rod is misaligned by more than 15 inches with its bank, or (b) if the rod cannot be moved by its drive mechanism, or (c) if its rod drop time is not met.

3.10.4.2 No more than one inoperable control rod shall be permitted during power operation.

3.10.4.3 If a control rod cannot be moved by its mechanism, boron concentration shall be changed to compensate for the withdrawn worth of the inoperable rod such that a shutdown margin equal to or greater than shown of Figure 3.10-3 results.

3.10.4.4 Power operation with an inoperable control rod shall not be allowed if the inoperable rod has a potential reactivity insertion upon ejection greater than 0.3% $\Delta k/k$ at rated power. The potential reactivity insertion of the inoperable rod shall be confirmed within 4 weeks to be less than 0.3% $\Delta k/k$ at rated power.

3.10.5 Rod Position Monitor

a. 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.

3.10.6 Core Exit Thermocouples

3.10.6.1 At least 4 core exit thermocouples per quadrant shall be available for readout when the reactor is critical.

3.10.6.2 Power operation may continue for 7 days with less than 4 core exit thermocouples per quadrant available for read out provided the licensee utilizes his best efforts to effect repairs and restore the number of operable thermocouples to that specified in 3.10.6.1 above. If, however, it is found that the inoperable components cannot be repaired, power operation is authorized with less than the requirements of 3.10.6.1 until the next refueling outage, provided during this period, the frequency of core survey using the movable in-core detectors is increased to once a week of power operation.

Basis:

The reactivity control concept is that reactivity changes accompanying changes in reactor power are compensated by control rod motion. Reactivity changes associated with xenon, samarium, fuel depletion, and large changes in reactor coolant temperature (operating temperature to cold shutdown) are compensated by changes in the soluble boron concentration. During power operation, the shutdown groups are fully withdrawn and control of reactor power is by the control groups. A reactor trip occurring during power operation will put the reactor into the hot shutdown condition.

The control rod insertion limits provide for achieving hot shutdown by reactor trip at any time, assuming the highest worth control rod remains fully withdrawn, with sufficient margins to meet the assumptions used in the accident analysis.[1] In addition, they provide a limit on the maximum inserted rod worth in the unlikely event of a hypothetical rod ejection, and provide for acceptable nuclear peaking factors. The lines shown on Figures 3.10-1 and 3.10-2 meet the shutdown requirement for the first and subsequent cycles. The maximum shutdown margin requirement occurs at end of cycle life and is based on the value used in analysis of the hypothetical steam break accident. Early in cycle life, less shutdown margin is required, and Figure 3.10-3 shows the shutdown margin equivalent to 1.95% reactivity at end-of-life with respect to an uncontrolled cooldown. All other accident analyses are based on 1% reactivity shutdown margin.

Positioning of the part-length rods is governed by the requirement to maintain the axial power shape within specified limits or to accept an automatic cutback of the overpower and overtemperature ΔT set points (see Specification 2.3). Thus, there is no need for imposing a limit on the physical positioning of the part-length rods.

The various control rod banks (shutdown banks, control banks A, B, C, D, and part-length rods are each to be moved as a bank, that is, with all rods in the bank within one step (5/8 inch) of the bank position. Position indication is provided by two methods: a digital count of actuation pulses which shows the demand position of the banks and a linear position indicator

(LVDT) which indicates the actual rod position.[2] The 15-inch permissible misalignment provides an enforceable limit below which design distribution is not exceeded. In the event that an LVDT is not in service, the effects of a malpositioned control rod are observable on nuclear and process information displayed in the control room and by core thermocouples and in-core moveable detectors. Determination of the core hot channel factors can be obtained by the operator, through hand analysis utilizing the in-core movable detector system in conjunction with thermocouples and primary loop temperature measurements. Standard procedures are followed in the determination of the core hot channel factors. Two hours is acceptable since complete rod misalignment (part-length or full-length control rod 12 feet out of alignment with its bank) does not result in exceeding core limits in steady-state operation at rated power. If the condition cannot be readily corrected, the specified reduction in power of 2% for each 1% the quadrant tilt is in excess of T%, will insure that design margins to core limits will be maintained under both steady-state and anticipated transient conditions.

A quadrant to average power tilt will be indicated by the ex-core detectors. The ex-core current tilt is indicated by the arrangement of the current recorders on the control board. Four 2-pen recorders are provided, the pens are grouped so that, in the absence of a tilt, the two ink traces coincide. Any divergency in the traces indicates a power tilt. Furthermore, a maximum-to-average alarm is provided for the upper and lower sets of ex-core currents. A power tilt ratio of T% as defined in 3.10.2.3 will insure that the rod power limit dictated by the loss-of-coolant accident, that accident for which there is the least margin with respect to operation at rated power, is not exceeded.

If, instead of determining the hot channel factors, the operator chooses simply to reduce power, the specified limit of 2% for each 1% the quadrant tilt exceeds T% maintains the minimum design margin to core safety limits for up to a 1.15 power tilt ratio. Resetting of the overpower trip setpoints ensures that the protection system basis is maintained for sustained plant operation. A tilt ratio of 1.15 or more is indicative of a serious performance anomaly and a plant shutdown is prudent.

The specified rod drop time is consistent with safety analyses that have been performed[1].

An inoperable rod imposes additional demands on the operator. The permissible number of inoperable control rods is limited to one in order to limit the magnitude of the operating burden, but such a failure would not prevent dropping of the operable rods upon reactor trip.

The reactivity worth limit for an inoperable control rod is consistent with the value found tolerable in the analysis of the hypothetical rod ejection accident[3].

The reactivity worth of any single inserted control rod is less than 0.2% Δk. This value will be confirmed by suitable tests of representative rods as part of the physics tests (FSAR, Table 13.3-1). The limit of 0.3% Δk therefore is only a restriction on plant operation with an inoperable control rod, if the long term non-uniform fuel depletion, due to the presence of the inoperable rod, result in an increase in the worth of the rod. The time limit of 4 weeks to confirm the worth of the inoperable control rod is short enough to prevent the rod worth from exceeding 0.3% Δk.

Core protection is provided by the Reactor Protection System, which utilizes the excore nuclear detector system to prevent core safety limits from being exceeded. No protective functions rely on the core exit thermocouples or the movable incore instrumentation system. These two core monitoring systems provided more detailed information which can be useful in the event the excore nuclear system indicates the presence of an abnormal situation in the core. The purpose of specification 3.10.6 is to emphasize the need to maintain these two systems operational and to focus the operator attention on the usefulness of the additional information which may be obtained by any one of these two systems in case an abnormal situation should develop.

References

(1) FSAR - Section 14

(2) FSAR - Section 7.3

(3) FSAR - Section 14.2.6

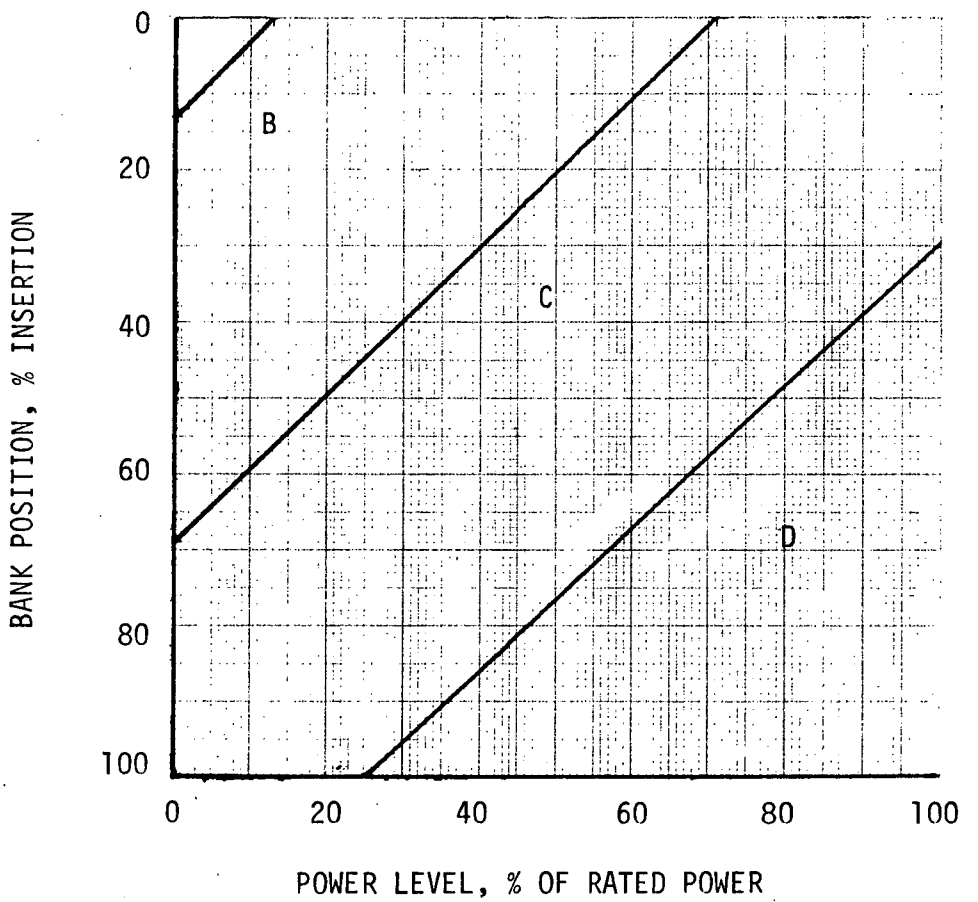

FIGURE 3.10-1 CONTROL BANK INSERTION LIMITS
FOR 4 LOOP OPERATION

Revised 3-22-73

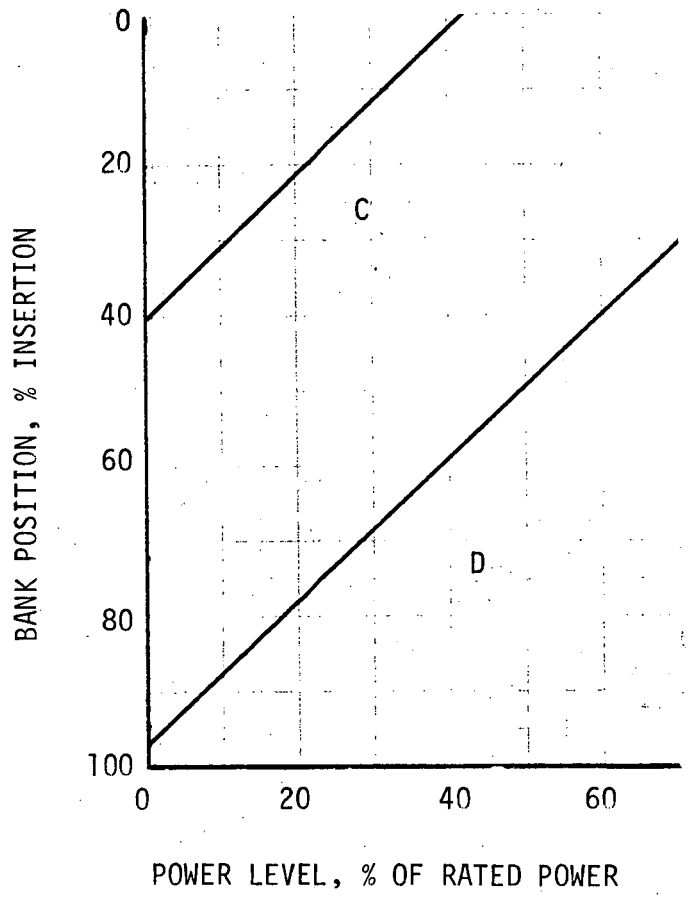

FIGURE 3.10-2 CONTROL BANK INSERTION LIMITS
FOR 3 LOOP OPERATION

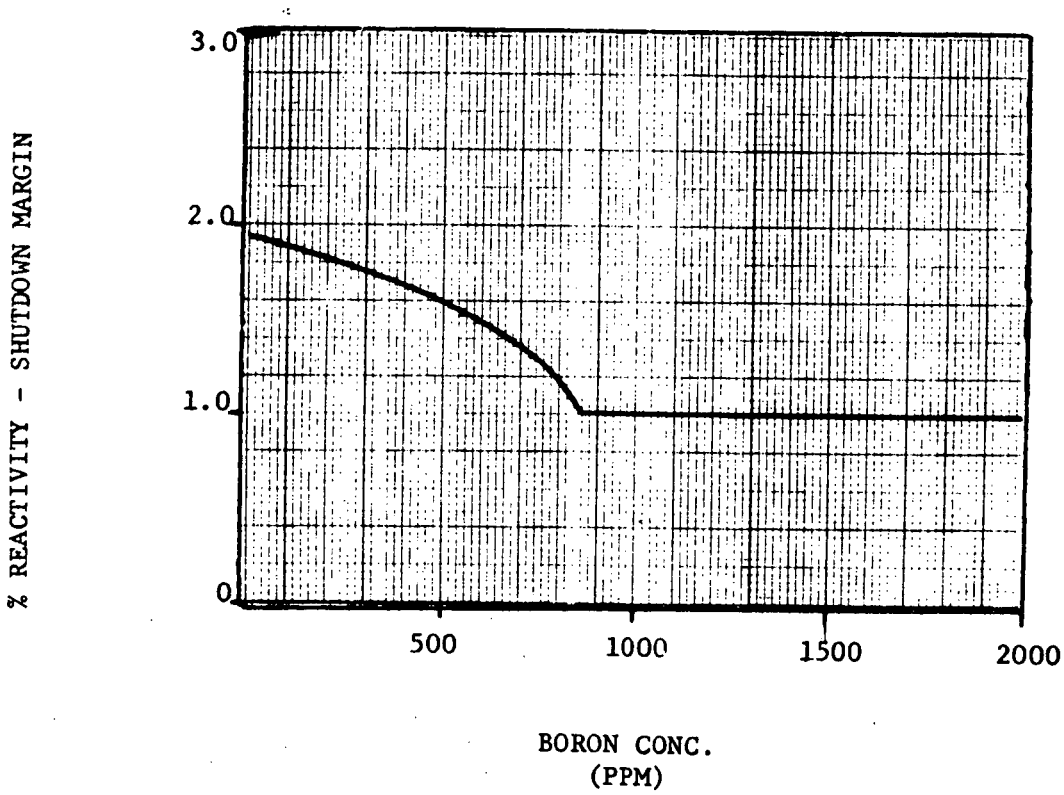

Figure 3.10-3

REQUIRED HOT SHUTDOWN MARGIN
vs
REACTOR COOLANT BORON CONCENTRATION

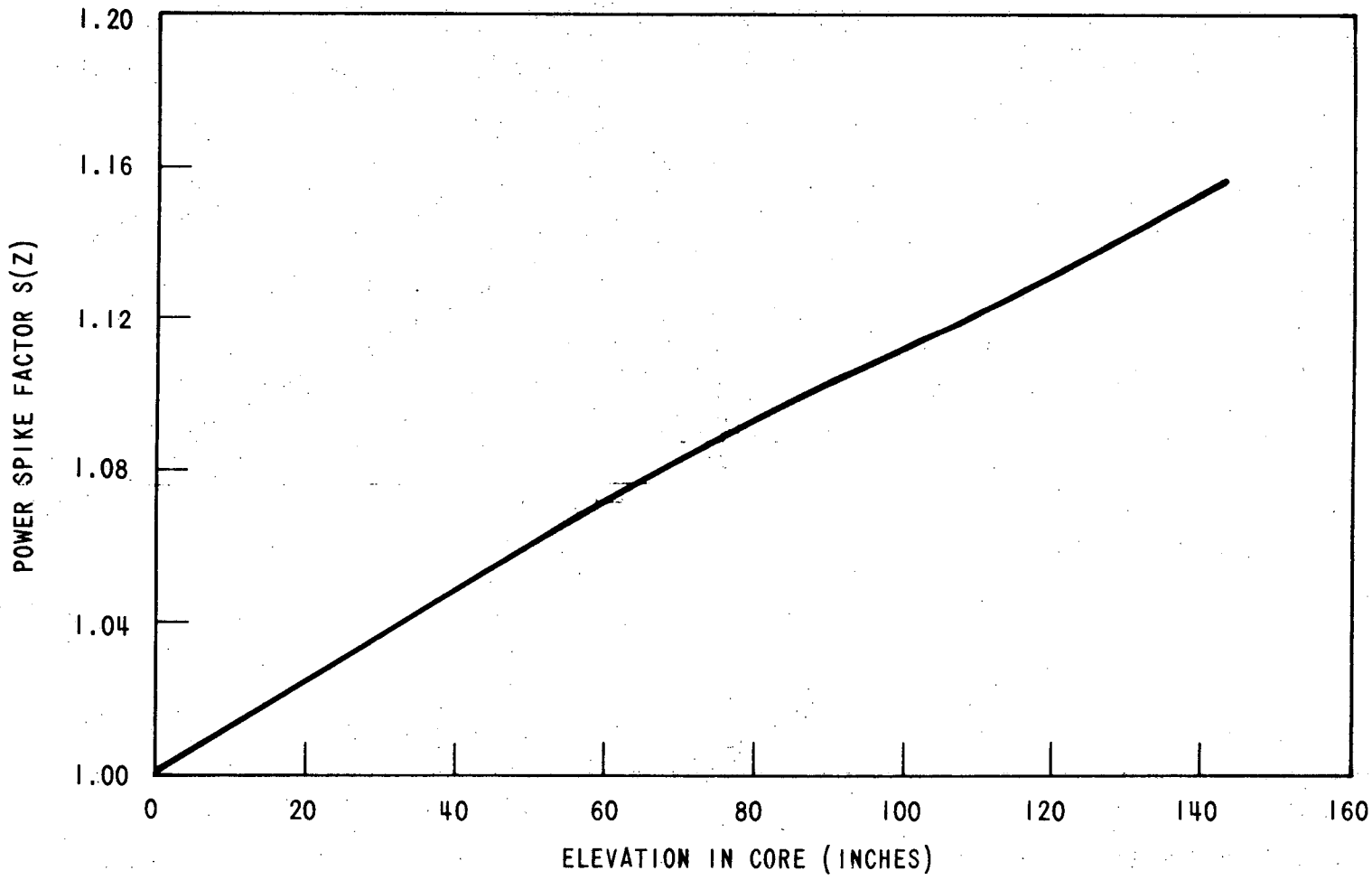

Figure 3.10-4. Power Spike Factor versus Elevation IPP2, Cycle 1

6288-1

3.11 MOVABLE IN-CORE INSTRUMENTATION

Applicability

Applies to the operability of the movable detector instrumentation system.

Objective

To specify functional requirements on the use of the in-core instrumentation system, for the recalibration of the excore axial off-set detection system.

Specification

A. A minimum of 2 thimbles per quadrant and sufficient movable in-core detectors shall be operable during re-calibration of the excore axial off-set detection system.

B. Power shall be limited to 90% of rated power for 4 loop or 65% of rated power for 3 loop operation if re-calibration requirements for excore axial off-set detection system, identified in Table 4.1-1, are not met.

Basis

The Movable In-core Instrumentation System[1] has six drives, six detectors, and 50 thimbles in the core. Each detector can be routed to sixteen or more thimbles. Consequently, the full system has a great deal more capability than would be needed for the calibration of the ex-core detectors.

To calibrate the excore detectors system, it is only necessary that the Movable In-core System be used to determine the gross power distribution in the core as indicated by the power balance between the top and bottom halves of the core.

After the excore system is calibrated initially, recalibration is needed only infrequently to compensate for changes in the core, due for example to fuel depletion, and for changes in the detectors.

If the recalibration is not performed, the mandated power reduction assures safe operation of the reactor since it will compensate for an error of 10% in the excore protection system. Experience at Beznau No. 1 and R. E. Ginna plants has shown that drift due to changes in the core or instrument channels is very slight. Thus the 10% reduction is considered to be very conservative.

Reference

(1) FSAR - Section 7.4

4 SURVEILLANCE REQUIREMENTS

4.1 OPERATIONAL SAFETY REVIEW

Applicability

Applies to items directly related to safety limits and limiting conditions for operation.

Objective

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

Specification

A. Calibration, testing, and checking of analog channel and testing of logic channel shall be performed as specified in Table 4.1-1.

B. Sampling and equipment tests shall be conducted as specified in Table 4.1-2 and 4.1-3, respectively.

Basis

A. 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 and steam system instrumentation.

Calibration

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

The nuclear flux (linear level) channels are calibrated daily against a heat balance standard to account for errors induced by changing rod patterns and core physics parameters.

Other channels are subject only to the "drift" errors induced within the instrumentation itself and, consequently, can tolerate longer 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 each refueling shutdown.

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

Thus, minimum calibration frequencies of once-per-day for the nuclear flux (linear level) channels, and once each refueling shutdown for the process system channels is considered acceptable.

Testing

The minimum testing frequency for those instrument channels connected to the safety system is based on an average unsafe failure rate of

2.5×10^{-6} failure/hrs. per channel. This is based on operating experience at conventional and nuclear plants. An unsafe failure is defined as one which negates channel operability and which, due to its nature, is revealed only when the channel is tested or attempts to respond to a bona fide signal.

For a specified test interval W and an M out of N redundant system with identical and independent channels having a constant failure rate λ, the average availability A is given by:

$$A = \frac{W - Q\left(\frac{W}{N-M+2}\right)}{W} = 1 - \frac{N!}{(N-M+2)!\,(M-1)!}(\lambda W)^{N-M+1}$$

where A is defined as the fraction of time during which the system is functional, and Q is the probability of failure of such a system during a time interval W.

For a 2-out-of-3 system A = 0.9999968, assuming a channel failure rate, λ, equal to 2.5×10^{-6} hr^{-1} and a test interval, W, equal to 720 hrs.

This average availability of the 2-out-of-3 system is high, hence the test interval of one month is acceptable.

Because of their greater degree of redundancy, the 1/3 and 2/4 logic arrays provide an even greater measure of protection and are thereby acceptable for the same testing interval. Those items specified for monthly testing are associated with process components where other means of verification provide additional assurance that the channel is operable, thereby requiring less frequent testing.

TABLE 4.1-1

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTS OF INSTRUMENT CHANNELS

	Channel Description	Check	Calibrate	Test	Remarks
1.	Nuclear Power Range	S	D (1) M* (3)	M (2)	1) Heat balance calibration 2) Signal to ΔT; bistable action (permissive, rod stop, trips) 3) Upper and lower chambers for axial off-set
2.	Nuclear Intermediate Range	S (1)	N.A.	P (2)	1) Once/shift when in service 2) Log level; bistable action (permissive, rod stop, trip)
3.	Nuclear Source Range	S (1)	N.A.	P (2)	1) Once/shift when in service 2) Bistable action (alarm, trip)
4.	Reactor Coolant Temperature	S	R	M (1) (2)	1) Overtemperature-ΔT 2) Overpower-ΔT
5.	Reactor Coolant Flow	S	R	M	
6.	Pressurizer Water Level	S	R	M	
7.	Pressurizer Pressure(High and Low)	S	R	M	
8.	6.9 Kv Voltage & Frequency	N.A.	R	M	Reactor protection circuits only
9.	Analog Rod Position	S	R	M	

* By means of the moveable incore detector system

TABLE 4.1-1 (CONTINUED)

	Channel Description	Check	Calibrate	Test	Remarks
10.	Rod Position Bank Counters	S	N.A.	N.A.	With analog rod position
11.	Steam Generator Level	S	R	M	
12.	Charging Flow	N.A.	R	N.A	
13.	Residual Heat Removal Pump Flow	N.A.	R	N.A.	
14.	Boric Acid Tank Level	W	R	N.A.	Bubbler tube rodded during calibration
15.	Refueling Water Storage Tank Level	W	R	N.A.	
16.	Boron Injection Tank Level	W	R	R	
17.	Volume Control Tank Level	N.A.	R	N.A.	
18.	(a) Containment Pressure	D	R	M	Wide range
	(b) Containment Pressure	S	R	M	Narrow range
19.	Process and Area Radiation Monitoring Systems	D	R	M	
20.	Boric Acid Make-up Flow Channel	N.A.	R	N.A.	
21.	Containment and Recirculation Sump Level	N.A.	N.A.	R	
22.	Accumulator Level and Pressure	S	R	N.A.	
23.	Steam Line Pressure	S	R	M	

TABLE 4.1-1 (CONTINUED)

	Channel Description	Check	Calibrate	Test	Remarks
24.	Turbine First Stage Pressure	S	R	M	
25.	Environmental Radiation Monitors	W	S.A.	N.A.	Calibrate by checking air flow meter on particulate monitor
26.	Logic Channel Testing	N.A.	N.A.	M	
27.	Turbine Overspeed Protection Trip Channel (Electrical)	N.A.	R	M	
28.	Control Room Ventilation	N.A.	N.A.	R	Check damper operation for accident mode with isolation signal

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

S - Each Shift
D - Daily
W - Weekly
M - Monthly
Q - Quarterly
S.A. - Semi-annually
P - Prior to each startup if not done previous week
R - Each Refueling Shutdown, but not to exceed 18 months, except for the first fuel cycle.

NA - Not applicable

TABLE 4.1-2

FREQUENCIES FOR SAMPLING TESTS

		Check	Frequency	Maximum Time Between Tests
1.	Reactor Coolant Samples	Gross Activity (1)	5 days/week (1)	3 days
		Radiochemical (2)	Monthly	45 days
		$\bar{E}$ Determination	Semiannually (3)	30 weeks
		Tritium Activity	Weekly (1)	10 days
		F, Cl & O_2	Weekly	10 days
2.	Reactor Coolant Boron	Boron Concentration	Twice/week	5 days
3.	Refueling Water Storage Tank Water Sample	Boron Concentration	Monthly	45 days
4.	Boric Acid Tank	Boron concentration	Twice/week	5 days
5.	Boron Injection Tank	Boron concentration	Monthly	45 days
6.	Spray Additive Tank	NaOH concentration	Monthly	45 days
7.	Accumulator	Boron concentration	Monthly	45 days
8.	Spent Fuel Pit	Boron concentration	Prior to Refueling	NA*
9.	Secondary Coolant	Iodine-131	Weekly (4)	10 days
10.	Liquid Radwaste Discharge Line Monitor	Gross β and γ Activity	Continuous (5)	N.A.
11.	Composite Discharge Canal Sampler	Radioactivity Analysis	Continuous Composite (6)	N.A.
12.	Liquid Radwaste Monitor Tanks	Radioactivity Analysis	Prior to each batch release	N.A.
13.	Stack Gas Monitor	Gross β and γ Activity	Continuous (5)	N.A.
14.	Iodine-Particulate Stack Gas Monitor	Iodine 131 and Particulate Activity	Continuous (5)	N.A.
15.	Containment Iodine-Particulate Monitor or Gas Monitor	Iodine 131 and particulate activity or gross gaseous activity	Continuous when operating at power (7)	N.A.

(1) A gross activity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units of μCi/cc, and when activity levels exceed 10% of limits specified in 3.9.b.1 and 3.9.c.1. The sampling frequency shall be increased to a minimum of once each day.

TABLE 4.1-2 (Cont'd)

FREQUENCIES FOR SAMPLING TESTS

(2) A radiochemical analysis shall consist of the quantitative measurement of each radionuclide with half life greater than 30 minutes making up at least 95% of the total activity of the primary coolant.

(3) $\bar{E}$ determination will be started when the gross analysis indicates $\geq$ 10 μ Ci/cc and will redetermined if the primary coolant gross radioactivity changes by more than 10 μ Ci/cc in accordance with Specification 3.1.D.

(4) When the iodine-131 activity exceeds 10% of the limit in Specification 3.4.A, the sampling frequency shall be increased to a minimum of once each day.

(5) Except as indicated in specification 3.9.

(6) Only when releasing radioactive effluents; sampling once/shift when discharging may be substituted when monitors are inoperable.

(7) Except as indicated in Specification 3.1.F.5.

*NA - Not Applicable

TABLE 4.1-3

FREQUENCIES FOR EQUIPMENT TESTS

		Check	Frequency	Maximum Time Between Tests
1.	Control Rods	Rod drop times of all full length rods	Each refueling shutdown	18 months**
2.	Control Rod	Partial movement of all full length rods	Every 2 weeks during reactor critical operations	20 days
3.	Pressurizer Safety Valves	Set point	Each refueling shutdown	18 months**
4.	Main Steam Safety Valves	Set point	Each refueling shutdown	18 months**
5.	Containment Isolation System	Automatic Actuation	Each refueling shutdown	18 months**
6.	Refueling System Interlocks	Functioning	Prior to each refueling shutdown	NA*
7.	Fire Protection System and Power Supply	Functioning	Annually	18 months
8.	Primary System Leakage	Evaluate	5 days/week	NA*
9.	Diesel Fuel Supply	Fuel Inventory	Weekly	10 days
10.	Turbine Steam Stop, Control Valves	Closure	Monthly	45 days
11.	Cable Tunnel Ventilation Fans	Functioning	Monthly	45 days
12.	Control Room and Fuel Handling Building Filtration System	Charcoal Filter Pressure Drop Test < 5 inches of water visual inspection Freon - 112 (or equivalent) test ≥ 99.5% at ambient conditions	Prior to each refueling outage***	18 months**

TABLE 4.1-3 (CONTINUED)

FREQUENCIES FOR EQUIPMENT TESTS

	Check	Frequency	Maximum Time Between Tests
13. Containment Air Filtration System	Visual Inspection	Every six months for the first two years and every refueling thereafter***	9 months (18 months)
	Pressure Drop Test < 5 inches of water	Each refueling***	18 months**
	Charcoal coupons: iodine and ignition temperature 50% removal for methyl iodine, no ignition below 300° C.	Every six months for the first two years and every refueling thereafter	9 months (18 months)
	HEPA filters DOP ≥ 99% efficiency	Each refueling***	18 months**

* NA - Not Applicable
** Except for the first fuel cycle
*** Or at any time work on the filters could alter their integrity

4.2 PRIMARY SYSTEM SURVEILLANCE

Applicability

Applies to pre-operational and in-service structural surveillance of the reactor vessel and primary system boundary.

Objective

To assure the continued integrity of the primary system boundary.

Specification

a. Prior to initial plant operation, a survey, using ultrasonic, visual and surface techniques, shall be made to establish pre-operational system integrity and establish baseline data.

b. Post-operational non-destructive inspections listed in Table 4.2.1 shall be performed as specified. The results obtained from compliance with this specification shall be evaluated after five years and the conclusions of this evaluation shall be reviewed with the AEC.

c. The structural integrity of the primary system boundary shall be maintained at the level required by the original acceptance standards throughout the life of the plant. Any evidence as a result of the inspections listed in Table 4.2.1, that defects have initiated or grown shall be investigated, including evaluation of comparable areas of the primary system.

d. Detailed records of each inspection shall be maintained to allow comparison and evaluation of future inspections.

4.2.2

The inspection interval shall be ten years.

4.2.3

The following definitions shall apply to the inspection methods employed in Table 4.2-1. The paragraphs referenced are corresponding paragraphs of Section XI of the ASME Code for In-Service Inspection of Nuclear Reactor Coolant Systems dated January 1970.

a. UT - Ultrasonic examination per paragraph IS 213.2.

b. RT - Radiographic examination per paragraph IS 213.1. Ultrasonic testing is an acceptable alternate for RT.

c. MT - Magnetic particle examination per paragraph IS 212.1.

d. PT - Liquid penetrant examination per paragraph IS 212.2.

e. V - Visual examination per paragraphs IS 211.1 or IS 211.2.

4.2.4

Examinations which reveal unacceptable structural defects in a category shall be extended to include an additional number (or areas) of system components or piping in the same category approximately equal to that initially examined. In the event further unacceptable structural defects are revealed, all remaining system components or piping in the category shall be examined to the extent specified in that examination category.

4.2.5

With the exception of those components or areas for which the examination may be deferred to the end of the inspection interval, at least 25 percent of the required examinations shall have been completed by the expiration of one-third of the inspection interval (with credit for no more than 33-1/3 percent if additional examinations are completed) and at least 50 percent shall have been completed by the expiration of two-thirds of the inspection

interval (with credit for no more than 66-2/3 percent). The remaining required examinations shall be completed by the end of the inspection interval. Successive inspections shall meet the requirements of Paragraph ISI-243 of the ASME Rules for In-Service Inspection of Nuclear Reactor Coolant Systems.

Basis

The inspection program, where practical, is in compliance with Section XI of the ASME Code for In-Service Inspection of Nuclear Reactor Coolant Systems dated January 1970. Though examinations in certain areas are desirable, it should be recognized that equipment and techniques to perform the inspection are still in development. In all areas scheduled for volumetric examination, a detailed pre-service mapping will be conducted using techniques anticipated to be used for post-operation examinations. The areas indicated for inspection represent those of representative stress levels, and therefore will serve to indicate potentical problems before significant flaws develop there or at other areas. As more experience is gained in operation of pressurized-water reactors, the time schedule and location of inspection may be altered or, should new techniques be developed, consideration may be given to incorporate these new techniques into this inspection program.

The use of conventional non-destructive test techniques can be applied to the inspection of most primary loop components except the reactor vessel. The reactor vessel presents special problems because of the radiation levels and the requirement for remote underwater accessibility to this component. Because of these limitations on access to the reactor vessel, serveral steps[1] have been incorporated into the design and manufacturing procedures in preparation for non-destructive test techniques which may be available in the future.

The techniques for inspection include visual inspections, ultrasonic, radiographic, magnetic particle and liquid penetrant testing of selected parts during refueling periods or other appropriate plant outages.

The inspection requirements of this section shall apply to all pressure-containing components that are part of the system boundary defined herein.

The system boundary includes all pressure vessels, piping, pumps and valves which are:

a. part of the reactor coolant system[2] or

b. connected to reactor coolant system, up to and including any and all of the following:

   (1) the outermost containment isolation valve[3,4] in system piping which penetrates primary reactor containment.

   (2) the second of two valves[5] normally closed during normal reactor operation in system piping which does not penetrate primary reactor containment and

   (3) the reactor coolant system safety and relief valves.

Exclusions

(1) Sample and instrumentation piping and socket-welded piping two inches and smaller.

(2) Components that can be isolated from the reactor coolant system by two valves (both closed, both open or one closed and the other open). Each valve must be capable of automatic actuation and its closure time must be such that, for postulated failure of the component during normal reactor operation (and the other valve is open), the reactor can be shut down and cooled down in an orderly manner assuming makeup is provided by the reactor coolant makeup system[5] only.

The examinations scheduled are listed in Table 4.2.1 and are discussed below:

A. Reactor Vessel and Closure Head

ITEM 1.1 (CATEGORY A) - Pressure-Containing Welds in Reactor Vessel Belt-Line Region

Due to the Indian Point Unit No. 2 plant design, the welds in the reactor vessel are not accessible from the O.D. It is intended that these welds be volumetrically examined from the I.D. when required, using remote, mechanized techniques. At present, there is no equipment available which will perform this examination. The examinations scheduled in Table 4.2-1 are predicated on the development of appropriate equipment. Since the examination of these welds requires removal of the core internals and thermal shield, the examinations are scheduled near the end of the ten-year inspection interval.

When the longitudinal and circumferential welds have received an exposure to neutron fluence in excess of 10^{19} nvt (E_n of 1 MeV or above), the length of weld in the high fluence region to be examined shall be increased to, at least, 50 percent.

ITEM 1.2 (CATEGORY B) - Longitudinal and Circumferential Welds in Shell (Other than those of Category A and C) and Meriodional and Circumferential Seam Welds in Bottom Head and Closure Head (Other than those of Category C)

Due to the Indian Point Unit No. 2 plant design, the welds in the reactor vessel are not accessible from the O.D. It is intended that these welds be volumetrically examined from the I.D., using remote, mechanized techniques. As discussed in Item 1.1 above, these examinations are predicated on the development of appropriate equipment.

A small portion of the welds between the head flange weld and the CRD shroud are accessible for an ultrasonic examination from the O.D. when the head is removed from the vessel. These welds will be examinations scheduled to be performed on these welds are shown in Table 4.2-1.

ITEM 1.3 (CATEGORY C) - Vessel-to-Flange and Head-to-Flange Circumferential Welds

The head flange weld can be examined using either mechanized or manual ultrasonic techniques. This weld is accessible when the head is removed for refueling.

Due to the Indian Point Unit No. 2 plant design, the vessel to flange weld in the reactor vessel is not accessible from the O.D. It is intended that this weld be volumetrically examined from the I.D. using remote mechanized techniques. As discussed in Item 1.1 above, the examinations are predicted on the development of appropriate equipment.

The examinations scheduled to be performed on these welds are shown in Table 4.2-1.

ITEM 1.4 (CATEGORY D) - Primary Nozzle-to-Vessel Welds and Nozzle-to-Vessel Inside Radiused Section

Due to the plant design, the vessel nozzle welds are accessible only from the I.D. It is believed that the inner radius of outlet nozzles can be examined without removing the core barrel. However, the core barrel must be removed to examine the inlet nozzles. For this reason, it is planned that the examination of the outlet nozzles be performed during the planned refueling outages near the third and sixth year and the inspection of the inlet nozzles near the end of the ten (10) year inspection interval.

At present, there is no equipment available to inspect these welds remotely. Thus, the examinations shown in Table 4.2-1 are predicated on the development of suitable equipment.

ITEM 1.5 (CATEGORY E-1) - Vessel Penetrations, Including Control Rod Drive Penetrations and Control Rod Housing Pressure Boundary Welds

The penetrations in this category are the control rod drive penetrations in the upper head and the instrument penetrations in the lower head. Because of the design of the vessel penetrations and the pressure boundary weld, no meaningful volumetric examinations can be performed. Visual examinations will be performed as discussed in Item 1.6 below.

ITEM 1.6 (CATEGORY E-2) - Vessel Penetrations Including Control Rod Drive Penetrations and Control Rod Housing Pressure Boundary Welds

The control rod drive penetrations in the upper head and the instrument penetrations in the upper and lower head are included in this category. The penetrations in the upper head can be visually inspected for leakage during the system hydrostatic test as defined by Paragraph IS 521 of ASME Code Section XI at or near the end of the ten-year interval. The penetrations in the lower head will also be examined for leakage during this test.

ITEM 1.7 (CATEGORY F) - Primary Nozzles to Safe-end Welds

There are dissimilar metal welds between the carbon steel nozzle forgings and the reactor coolant piping. These welds will be ultrasonically tested during the inspection interval pending the development of suitable, remote ultrasonic equipment. This inspection will coincide with the Item 1.4 inspection if supporting equipment is developed. Limited access to the O.D. of these welds is provided by removable plugs in the primary shield above the nozzles and removable insulation covering the nozzle welds. Exception is taken to performing a surface examination on these welds due to anticipated radiation levels and physical access.

ITEMS 1.8; 1.9; 1.10 (CATEGORY G-1) - Closure Studs, Nuts, Washers, Bushings and Ligaments Between Threaded Stud Holes

The closure studs, nuts, washers, bushings and ligaments between threaded stud holes will be inspected in accordance with Section XI of the ASME Code. The examinations scheduled for this inspection are shown in Table 4.2-1.

ITEM 1.11 (CATEGORY G-2) - Closure Studs, Nuts, Washers, Busings and Ligaments Between Threaded Stud Holes

There are no pressure-retaining bolts less than two inches on the Indian Point Unit No. 2 vessel.

ITEM 1.12 (CATEGORY H) - Integrally-Welded Vessel Supports

There are a total of four vessel support pads welded to inlet and outlet nozzles on the Indian Point Unit No. 2 vessel. In accordance with Category H of Table IS-251 of the Code, the area to be examined should be the weld connection between the nozzle and the vessel shell. This examination is covered by Item 1.4 above.

ITEMS 1.13 and 1.14 (CATEGORY I-1) - Vessel Cladding

The cladding in the closure head can be visually examined and liquid penetrant examined when the head is removed from the vessel for refueling. Portions of the cladding in the reactor vessel are accessible for remote visual examinations through access ports in the core barrel support flange. The examinations scheduled to be performed are shown in Table 4.2-1.

ITEM 1.15 (CATEGORY N) - Interior Surfaces and Integrally-Welded Internal Supports

The internal surfaces and internal components of the reactor vessel will be inspected in accordance with Section XI of the Code.

ITEM 2.1 (CATEGORY B) - Longitudinal and Circumferential Welds

Examination of the pressurizer longitudinal and circumferential welds will be performed on accessible portions of the pressurizer shell. Approximately 50 percent of the shall is enclosed in a biological and missile shield and is, therefore, not accessible for examination.

ITEM 2.2 (CATEGORY D) - Nozzle-to-Vessel Welds

The nozzles on the pressurizer are cast with the upper and lower heads; therefore, no inspections are required for these items.

ITEM 2.3 (CATEGORY E-1) - Heater Connections

The heater connections on the I.D. of the pressurizer are not accessible for visual or surface examination. The external connections are accessible for a visual examination and will be inspected as discussed in Item 2.4 below.

ITEM 2.4 (CATEGORY E-2) - Heater Connections

The pressurizer heater external connections are accessible for visual examination. These connections will be visually examined for leakage during the system hydrostatic test as defined by Paragraph IS-521 of ASME Section XI at or near the end of the ten-year interval. The instrument and sample nozzles of the pressurizer are included in this inspection.

ITEM 2.5 (CATEGORY G-1) - Pressure Retaining Bolting

There is no pressure-retaining bolting on the Indian Point Unit No. 2 pressurizer two inches or greater in diameter.

ITEM 2.6 (CATEGORY G-2) - Pressure Retaining Bolting

The pressurizer manway bolting will be inspected in accordance with the requirements of Section XI.

ITEM 2.7 (CATEGORY H) - Integrally-Welded Vessel Supports

There are no integrally welded vessel supports on the Indian Point Unit No. 2 pressurizer.

ITEM 2.8 (CATEGORY I-2) - Vessel Cladding

There will be a visual examination of the vessel cladding in accordance with ASME Section XI.

B. Steam Generator

ITEM 3.1 (CATEGORY B) - Longitudinal and Circumferential Welds, Including Tubesheet-to-Head or Shell Welds on the Primary Side

The primary head for the Indian Point Unit No. 2 steam generators is a one-piece casting. Thus, the only weld covered by this category is the tubesheet-to-head weld. It is believed that this weld can be examined from the O.D. of the steam generator by ultrasonic techniques. The examinations scheduled for this weld are shown in Table 4.2-1.

ITEM 3.2 (CATEGORY D) - Primary Nozzle-to-Vessel Head Welds and Nozzle-to-Head Inside Radiused Section

The primary nozzles are cast with the primary head; therefore, no inspections are planned for this item.

ITEM 3.3 (CATEGORY F) - Primary Nozzle to Safe-End Welds

The steam generator safe-ends are a buttered end preparation of the cast nozzle and are located between the nozzle and cast fittings. It

is not anticipated that a meaningful ultrasonic examination of these welds can be performed. However, these examinations are being conducted presently and if meaningful results are obtained, these areas shall be inspected per ASME Section XI requirements.

ITEM 3.4 (CATEGORY C-1) - Pressure-Retaining Bolting

There is no pressure-retaining bolting two inches or greater in diameter on the steam generator.

ITEM 3.5 (CATEGORY G-2) - Pressure-Retaining Bolting

The pressure-retaining bolting on the steam generator primary manway will be inspected in accordance with the requirements of Section XI.

ITEM 3.6 (CATEGORY H) - Integrally-Welded Vessel Supports

Indian Point Unit No. 2 generator supports are not integrally welded to the steam generator. Thus, this item does not apply.

ITEM 3.7 (CATEGORY 1-2) - Vessel Cladding

The interior of the primary side of the steam generator is accessible through a manway in each side of the primary head. One patch of cladding (36 square inches) in each side of the primary head will be visually examined during the inspection interval. The examinations scheduled are listed in Table 4.2-1.

C. Piping Pressure Boundary

ITEM 4.1 (CATEGORY F) - Vessel, Pump and Valve Safe-Ends to Primary Pipe Welds and Safe-Ends in Branch Piping Welds

There are no pump or valve safe-ends in the primary system boundary. The examinations scheduled to be performed on these welds are shown in Table 4.2-1.

ITEM 4.2 (CATEGORY J) - Circumferential and Longitudinal Pipe Welds

Due to the design of the Indian Point Unit No. 2 piping systems, there may be areas where access to piping welds will be impossible due to high radiation levels and/or physical access problems. Exception is taken to performing inspections in these areas. Exception is also taken to performing inspections on socket welds within the primary boundary and sampling and instrumentation piping and thermowells.

The remaining welds in the primary system will be examined in such a manner as to cumulatively cover 25% of the welds during the inspection interval.

The examinations scheduled are given in Table 4.2-1.

ITEM 4.3 (CATEGORY G-1) - Pressure-Retaining Bolting

The only pressure-retaining bolting in the piping boundary is at the upstream side of the pressurizer safety valve connections to the relief line. This bolding is less than two (2) inches in diameter and thus is covered by Item 4.4 below.

ITEM 4.4 (CATEGORY G-2) - Pressure-Retaining Bolting

The bolting connections joining the upstream side of the pressurizer safety valves to the relief line will be examined in accordance with Section XI of the ASME Code, as shown in Table 4.2-1.

ITEM 4.5 (CATEGORY K-1) - Integrally-Welded Supports

The accessible integrally-welded supports in the Indian Point Unit No. 2 piping systems subject to this inspection will be examined in accordance with Section XI of the Code. The examinations scheduled are shown in Table 4.2-1.

ITEM 4.6 (CATEGORY K-2) - Piping Support and Hanger

The accessible piping supports and hangers of the piping systems subject to this inspection will be examined in accordance with the Code. The examinations scheduled are shown in Table 4.2-1.

Pump Pressure Boundary

The only pumps subject to inspection are the reactor coolant pumps. The following items apply to these pumps.

ITEM 5.1 (CATEGORY L-1) - Pump Casing Welds

The reactor coolant pump casing is a weldment of two cast shells. At this time, there are no proven means of volumetrically inspecting the pump casing welds in service; therefore, no inspections are planned. However, the accessible internal surface of one pump casing weld shall be visually and liquid penetrant inspected. The pump casing weld inspected shall correspond to the pump casing inspected in Item 5.2.

ITEM 5.2 (CATEGORY L-2) - Pump Casing

The internal pressure boundary surfaces of the reactor coolant pumps are not accessible during normal or refueling outages. If removal of the pump internals is required during the inspection interval, there will be a visual examination of the internal surfaces of one disassembled pump. Otherwise, the examination of the internal surfaces of one disassembled pump will be performed at or near the end of the inspection interval.

ITEM 5.3 (CATEGORY F) - Nozzle-to-Safe-End Welds

There are no nozzle-to-safe-end welds on the Indian Point Unit No. 2 reactor coolant pumps.

ITEM 5.4 (CATEGORY G-1) - Pressure-Retaining Bolting

The reactor coolant pump main flange studs are greater than two (2) inches in diameter. These studs will be examined in accordance with the code. The examinations scheduled are shown in Table 4.2-1.

ITEM 5.5 (CATEGORY G-2) - Pressure-Retaining Bolting

There is pressure-retaining bolting less than two (2) inches in diameter, associated with the reactor coolant pump seals. Since this bolting is not normally accessible, examination of this bolting will be performed only when the pump is disassembled for maintenance purposes.

ITEM 5.6 (CATEGORY K-1) - Integrally-Welded Supports

The reactor coolant pump supports consist of a cast foot welded to the pump casing. There are no currently know techniques for ultrasonically inspecting these welds.

ITEM 5.7 (CATEGORY K-2) - Supports and Hangers

The reactor coolant pump supports will be visually examined during the inspection interval in accordance with Section XI of the code. The examinations scheduled are shown in Table 4.2-1.

Valve Pressure Boundary

The inspections in this category are limited to accessible valves three (3) inches or greater in the system boundary.

ITEM 6.1 (CATEGORY M-1) - Valve-Body Welds

None of the valves subject to this inspection have pressure containing body welds and thus, this item is not applicable.

ITEM 6.2 (CATEGORY M-2) - Valve Bodies

The internal surfaces of one disassembled valve (with or without pressure-containing welds) in each of the groups of valves of the same construction design, manufacturing method, manufacturer and performing similar functions in the system shall be examined during each inspection interval. The examination of the valve bodies may be performed on the same valves selected for volumetric examination of the pressure-containing welds.

ITEM 6.3 (CATEGORY F) - Valve-to-Safe-End Welds

There are no valve-to-safe-end welds in the piping boundary subject to this examination.

ITEM 6.4 (CATEGORY G-1) - Pressure-Retaining Bolting

There is no pressure-retaining bolting greater than two (2) inches in the valves subject to this examination.

ITEM 6.5 (CATEGORY G-2) - Pressure-Retaining Bolting

The bolting subject to this examination will be the bonnet bolting in valves three (3) inches in size or greater. This bolting will be inspected in accordance with Section XI of the code, as shown in Table 4.2-1.

ITEM 6.6 (CATEGORY K-1) - Integrally-Welded Supports

There are no integrally-welded supports on the valves subject to this examination.

ITEM 6.7 (CATEGORY K-2) - Supports and Hangers

The supports and hangers of the valves subject to this examination shall be visually examined in accordance with Section XI of the code, as shown in Table 4.2-1.

D. Miscellaneous Inspections

ITEM 7.1 - Primary Pump Flywheels

The flywheels shall be visually examined at the first refueling. At each subsequent refueling, one different flywheel shall be examined by ultrasonic methods. The examinations scheduled are shown in Table 4.2-1.

ITEM 7.2 - Materials Irradiation Surveillance Specimens

The reactor vessel surveillance program includes six specimen capsules to evaluate radiation damage based on pre-irradiation and post-irradiation testing of specimens. The specimens will be removed and examined at the following intervals.

Capsule 1 Relpacement of first region of core
Capsule 2 Replacement of second region of core
Capsule 3 Replacement of fourth region of core
Capsule 4 End of the tenth year of operation
Capsule 5 End of the fifteenth year of operation
Capsule 6 End of the twentieth year of operation

(1) FSAR Section 4.5

NOTES

(1) Examinations of certain reactor vessel areas such as longitudinal and circumferential shell welds, vessel to flange weld, primary nozzle to vessel welds etc., require the use of equipment and techniques which have not as yet been fully developed. In anticipation that such equipment and techniques will be developed within the next several years to the point of practical application, a pre-operational ultrasonic survey of these areas within the reactor vessel will be made to establish baseline data.

(2) The reactor coolant system is that system which contains primary reactor coolant at operating pressure during normal reactor operations.

(3) Containment isolation valves are those valves in system piping which penetrates the primary reactor containment and which can serve to isolate the system inside of containment from portions of the same system located outside of containment. Simple check valves are not acceptable for this purpose unless they are capable of automatic actuation upon an isolation signal.

(4) Two check valves in series are acceptable.

(5) See Note 1 on Page 4 of Section II.

TABLE 4.2-1

Item No.	Examination Category*	Components and Parts to be Examined	Method	Extent of Examination (Percent in 10 Year Interval)	Remarks
		REACTOR VESSEL AND CLOSURE HEAD			
1.1	A	Longitudinal and circumferential shell welds in core region (2)	UT	Longitudinal - 10% Circumferential - 5%	Those inspections are predicated on the development of remote mechanical ultrasonic examination devices.
1.2	B	Longitudinal and circumferential welds in shell (other than those of Category A and C), and meriodional and circumferential steam welds in bottom head and closure head (other than those of Category C)	UT	Longitudinal - 10% Circumferential - 5%	These inspections are predicated on the development of remote, mechanical ultrasonic examination devices.
1.3	C	Vessel-to-flange and head-to-flange circumferential welds	UT	100%	The vessel-to-flange weld inspection is predicated on the development of remote, mechanical ultrasonic examination devices.

* Refer to Table IS-251 in Section XI for definition of "Examination Categories"

TABLE 4.2-1 (Continued)

Item No.	Examination Category	Components and Parts to be Examined	Method	Extent of Examination (Percent in 10 Year Interval)	Remarks
1.4	D	Primary nozzle-to-vessel welds and nozzle-to-vessel inside radiused section	UT	100%	These inspections are predicated on the development of remote, mechanical ultrasonic examination devices.
1.5	E-1	Vessel penetrations, including control rod drive penetrations and control rod housing pressure boundary welds.		Not applicable.	Not applicable.
1.6	E-2	Vessel penetrations, including control rod drive penetrations and control rod housing pressure boundary welds.	V	25%	The examinations will be a visual examination for leakage during the system hydrostatic test at or near the end of the ten-year inspection interval.
1.7	F	Primary nozzles to safe-end welds	UT & V	100%	These UT inspections are predicated on the development of remote mechanical ultrasonic examination devices. The individual visual examination performed during each inspection shall cover 100% of the circumference of the safe-end welds. All safe-end welds shall be examined during the inspection interval.

TABLE 4.2-1 (Continued)

Item No.	Examination Category	Components and Parts to be Examined	Method	Extent of Examination (Percent in 10 Year Interval)	Remarks
1.8	G-1	Closure studs and nuts	UT & V or PT	100%	
1.9	G-1	Ligaments between threaded stud holes	UT	100%	
1.10	G-1	Closure washers, bushings	V	100%	
1.11	G-2	Pressure-retaining bolting		Not applicable	Not applicable
1.12	H	Integrally-welded vessel supports		See remarks	This inspection is covered by Item 1.4
1.13	I-1	Closure head cladding	PT & V	6 Patches	
1.14	I-1	Vessel cladding	V	6 Patches	
1.15	N	Interior surfaces and internals and integrally-welded internal supports	V	See remarks	The examination of interior vessel surfaces, internals, and the space below the reactor core, which are made accessible for examination by the removal of components during normal refueling

TABLE 4.2-1 (Continued)

Item No.	Examination Category	Components and Parts to be Examined	Method	Extent of Examination (Percent in 10 Year Interval)	Remarks
					outages shall be performed during each refueling period. Where access to the space below the reactor core during normal refueling outages precludes inspection of this space, at least one examination, at or near the end of each inspection interval, shall be conducted under conditions which enable inspection.
		PRESSURIZER			
2.1	B	Longitudinal and circumferential welds.	V & UT	Longitudinal - 10% Circumferential - 5%	Accessibility of welds is limited by biological and missile shield.
2.2	D	Nozzle-to-vessel welds	V & UT	See remarks	Instrument and sample nozzles are included in Item 2.4.
2.3	E-1	Heater connections		See remarks	These connections are considered in Item 2.4.

TABLE 4.2-1 (Continued)

Item No.	Examination Category	Components and Parts to be Examined	Method	Extent of Examination (Percent in 10 Year Interval)	Remarks
2.4	E-2	Heater connections and instrument and sample nozzles.	V	See remarks	Visual inspections for leakage will be performed on at least 25% of the penetrations during the system hydrostatic test.
2.5	G-1	Pressure-retaining bolting		Not applicable.	
2.6	G-2	Pressure-retaining bolting	V	100%	
2.7	H	Integrally-welded vessel supports		Not applicable	
2.8	I-2	Vessel cladding	V	1 Patch	One (1) patch (36 square inches) on the primary side near the manway will be examined during the ten-year interval.
2.9		Pressurizer Plate Inclusion Area	UT	See Remarks	UT of pressurizer plate at inclusion area during the preoperational test and during the first and second refueling. Should these examinations indicate no change in inclusion pattern, the applicant may subsequently follow the inservice inspection requirements of the ASME Section XI Code, provided this inclusion area is included as part of Category B examination Requirements (Table IS-251).

TABLE 4.2-1 (Continued)

Item No.	Examination Category	Components and Parts to be Examined	Method	Extent of Examination (Percent in 10 Year Interval)	Remarks
		HEAT EXCHANGERS (CLASS A) AND STEAM GENERATORS			
3.1	B	Longitudinal and circumferential welds, including tube-sheet-to head or shell welds on the primary side.	UT	5% See Remarks	The inspection is limited to the circumferential weld attaching the tube sheet to the lower head.
3.2	D	Primary nozzle-to-vessel head welds and nozzle-to-head inside radiused section		See Remarks	The primary nozzles are cast with the head. No inspections are planned.
3.3	F	Primary nozzle-to-safe-end welds	V & PT	100%	Not anticipated that meaningful UT results can be obtained.

TABLE 4.2-1 (Continued)

Item No.	Examination Category	Components and Parts to be Examined	Method	Extent of Examination (Percent in 10 Year Interval)	Remarks
3.4	G-1	Pressure-retaining bolting		Not applicable	
3.5	G-2	Pressure-retaining bolting	V	100%	
3.6	R	Integrally-welded vessel supports		Not applicable	
3.7	I-2	Vessel cladding	V	1 patch	One (1) patch (36 square inches) in each primary side will be examined during the ten-year interval.
3.8		Steam Generator No. 21; Shell Inclusion Area	UT	See remarks.	UT of Steam Generator No. 21 shell at inclusion area during shutdowns for refueling for the first ten years of operation. Should these examinations indicate no change in inclusion pattern, the inspections of the inclusion area may subsequently be decreased to at least once during each inspection interval.

TABLE 4.2-1 (Continued)

Item No.	Examination Category	Components and Parts to be Examined	Method	Extent of Examination (Percent in 10 Year Interval)	Remarks
		PIPING PRESSURE BOUNDARY			
4.1	F	Vessel, pump and valve safe-ends to primary pipe welds and safe-ends in branch piping welds.	Ut, PT & V	100%	This examination covers only the pressurizer safe-ends.
4.2	J	Circumferential and longitudinal pipe welds	V & UT	25%	(1) Exception is taken to inaccessible welds. (2) Exception is taken for socket welds. (3) Exception is taken for sampling and instrumental piping and thermowells.
4.3	G-1	Pressure-retaining bolting		Not applicable	
4.4	G-2	Pressure-retaining bolting	V	100%	

TABLE 4.2-1 (Continued)

Item No.	Examination Category	Components and Parts to be Examined	Method	Extent of Examination (Percent in 10 Year Interval)	Remarks
4.5	K-1	Integrally-welded supports	V & UT	100%	Exception is taken for supports which are not accessible.
4.6	K-2	Piping support and hangers	V	100%	Exception is taken for those supports which are not accessible.
		PUMP PRESSURE BOUNDARY			
5.1	L-1	Pump casing welds	V & PT	100%	No meaningful ultrasonic examinations can be performed on these welds.
5.2	L-2	Pump casings	V	See remarks	Examination will be made only when pump internals are removed for other reasons.
5.3	F	Nozzle-to-safe-end welds		Not applicable	
5.4	G-1	Pressure-retaining bolting	UT & V	100%	

TABLE 4.2-1 (Continued)

Item No.	Examination Category	Components and Parts to be Examined	Method	Extent of Examination (Percent in 10 Year Interval)	Remarks
5.5	G-2	Pressure-retaining bolting	V	See remarks	Bolting will be inspected only when pump is disassembled for other reasons, but at least once at or near the end of each inspection interval.
5.6	K-1	Integrally-welded supports	UT or PT and V	25%	
5.7	K-2	Supports and hangers	V	100%	
		VALVE PRESSURE BOUNDARY			
6.1	M-1	Valve-body welds		Not applicable	
6.2	M-2	Valve bodies		See remarks	Exception taken for valves which are not accessible or which are not disassembled for maintenance purposes during the inspection interval.
6.3	F	Valve-to-safe-end welds		Not applicable	

TABLE 4.2-1 (Continued)

Item No.	Examination Category	Components and Parts to be Examined	Method	Extent of Examination (Percent in 10 Year Interval)	Remarks
6.4	G-1	Pressure-retaining bolting		Not applicable	
6.5	G-2	Pressure-retaining bolting	UT & V	100%	Exception is taken for valves which are not accessible.
6.6	K-1	Integrally-welded supports		Not applicable	
6.7	K-2	Supports and hangers	V	100%	Exception is taken for supports and hangers which are not accessible.
7.1		Primary pump flywheel	V & UT	See remarks	The flywheels shall be visually examined at the first refueling. At each subsequent refueling, one different flywheel shall be examined by ultrasonic methods.
7.2		Irradiation Specimen Schedule	Tensile and Charpy V Notch (Wedge Open Loading)	See remarks	

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.

4.4 CONTAINMENT TESTS

Applicability

Applies to containment leakage.

Objective

To verify that potential leakage from the containment is maintained within acceptable values.

Specification

I. Integrated Leakage Rate Test - Pre-operational

A. Integrated Leakage Rate Tests

1. Integrated leakage rate tests shall be performed prior to initial plant operations at the containment design pressure (P_p) of 47 psig and at a reduced test pressure (P_t) of 23.5 psig to establish the respective measured leakage rates L_{pm} and L_{tm}.

2. The test duration shall not be less than 24 hours for integrated leakage rate measurements, and shall be extended a sufficient period of time to verify, by superimposing a known leak rate on the containment, the validity and accuracy of the leakage rate results.

3. The leakage test will be performed with the double penetration and weld channel zones open to the containment atmosphere.

4. The governing criterion for acceptance is that the maximum allowable post-accident leakage rate (L_a) shall not exceed 0.1 weight percent per day of containment steam-air atmosphere at 47 psig and 271°F, which are the maximum conditions of the design basis accident.

B. Sensitive Leakage Rate Test

1. A sensitive leakage rate test shall be conducted with the penetrations, weld channels and certain double gasketed seals and isolation valve interspaces at 50 psig and with the containment building at atmospheric pressure.

2. The test shall be considered satisfactory if the leak rate forthe double penetrations, weld channel and other pressurized zones is equal to or less than 0.2% of the containment free volume per day.

II <u>Integrated Leakage Rate Test - Postoperational</u>

A. Integrated Leakage Rate Test

1. The integrated leakage rate tests shall be performed at intervals specified in 4.4.II.A.6(a) at an initial pressure (beginning of test) at or above 23.5 psig (50% of design pressure).

2. The test duration shall not be less than 24 hours, and shall be extended a sufficient period of time to verify, by superimposing a known leak rate on the containment the validity and accuracy of the leakage rate results.

3. The test shall be performed without preliminary leak detection surveys or leak repairs. Leak repairs, if required during the integrated leakage rate test, shall be preceded whenever possible by local leakage rate measurements. The leakage rate difference, prior to and after repair and corrected to the test pressure (P_t) shall be indicated in the report of test results.

4. Closure of the containment isolation valves for the purpose of the test shall be accomplished by the means provided for normal operation of the valves.

5. Acceptance Criterion

(a) The governing criterion for acceptance is that the maximum allowable leakage rate, L_a, shall not exceed 0.10 weight percent per day of containment steam-air atmosphere at 47 PSIG (P_a) and 271°F (T_a) which are the maximum conditions of the design basis accident.

(b) The allowable operational leakage rate (L_{to}) which shall be met prior to resumption of power operation following a test (either as measured or following repairs and retest) shall not exceed .75 L_t, where L_t is defined by II.A.5(c).

(c) The allowable test leakage rate (L_t) at the reduced test pressure shall not exceed the lesser of L_a (L_{tm}/L_{pm}) or La $(P_t/P_p)^{1/2}$. The subscript m refers to values of the leakage measured during pre-operational tests. The subscripts p and t refer to tests at accident pressure and reduced pressure, respectively. Subscript a refers to accident conditions at accident pressure.

6. Frequency

(a) After the initial pre-operational leakage rate tests, an integrated leakage rate test at pressure P_t shall be performed approximately midway between the major shutdowns for inservice inspection conducted at 10 year intervals. In addition, an integrated leakage rate test at pressure P_t shall be performed at the end of the 10 year interval, coinciding with the inservice inspection shutdown.

B. Sensitive Leakage Rate Test

A sensitive leakage rate test as defined in Specification 4.4.I.B shall be performed during each shutdown for major fuel reloading.

III. Report of Test Results

Each integrated leakage rate test will be the subject of a summary technical report, and will include a summary of continuous leakage rate measurements as well.

IV. Continuous Leak Detection Testing via the Containment Penetration and Weld Channel Pressurization System

1. The upper limit for long-term uncorrected air consumption for the pressurization system shall be 0.2% of the containment volume per day (sum of four headers) at the system operating pressure, contingent on the following:

   a. Pressure in all pressurization zones is maintained above incident pressure.

   b. Air supply is maintained from the compressed air systems.

   c. The full complement of standby nitrogen cylinders is charged.

V. Corrective Action

1. If any time it is determined that the limit of IV.1 is exceeded, repairs shall be initiated immediately.

2. If repairs are not completed and conformance to the acceptance criterion is not demonstrated within 7 days, the reactor shall be shut down until repairs are effected and the continuous leakage meets the acceptance criterion.

VI. Isolation Valve Tests

Isolation valves shall be tested for operability at a frequency of at least every refueling. Isolation valves which are pressurized by the Penetration and Weld Channel Pressurization System will be leakage tested as part of the Sensitive Leakage Rate Test as described in 4.4.I.B. above.

VII. Residual Heat Removal System

A. Test

1. (a) The portion of the Residual Heat Removal System that is outside the containment shall be tested either by use in normal operation or hydrostatically tested at 350 psig at the interval specified in VII-D below.

   (b) Suction piping from the containment sump to the Residual Heat Removal System shall be hydrostatically tested at no less than 100 psig at the interval specified in VII-D below.

2. Visual inspection shall be made for excessive leakage during these tests from components of the system. Any significant leakage shall be measured by collection and weighing or by another equivalent method.

B. Acceptance Criterion

The maximum allowable leakage from the residual heat removal system components (located outside the containment) shall not exceed two gallons per hour.

C. Corrective Action

1. Repairs or isolation shall be made as required to maintain leakage within the acceptance criterion of VII-B.

2. If leakage is not reduced to the limits of VII-B during the subsequent weekend, the reactor shall be placed in the hot shutdown condition.

   If the conditions of VII-B cannot be met during an additional 48 hours, the reactor shall be placed in the cold shutdown condition.

D. Test Frequency

Tests of the residual heat removal system will be conducted at every refueling.

VIII. Annual Inspection

A detailed visual examination of the accessible interior and exterior surfaces of the containment structure and its components shall be performed annually and prior to any integrated leak test, to uncover any evidence of deterioration which may affect either the containment's sturctural integrity or leak-tightness. The discovery of any significant deterioration shall be accompanied by corrective actions in accord with acceptable procedures, non-destructive tests and inspections, and local testing where practical, prior to the conduct of any integrated leak test. Such repairs shall be reported as part of the test results.

IX. Containment Modifications

Any major modification or replacement of components of the containment performed after the initial preoperational leakage rate test shall be followed by either an integrated leakage rate test, or a local leak detection test and shall meet the acceptance criteria of II.A and I.B, respectively. Modifications or replacements performed directly prior to the conduct of an integrated leakage rate test shall not require a separate test.

Bases

The containment is designed for an accident pressure of 47 psig.[1] While the reactor is operating, the internal environment of the containment will be air at essentially atmospheric pressure and a maximum temperature of approximately 120°F. With these initial conditions, the temperature of the steam-air mixture at the peak accident pressure at 47 psig is 271°F.

Prior to initial operation, the containment will be strength tested at 54 psig and then will be leak-tested. The acceptance criterion for this pre-operational leakage rate test has been established as 0.1% per 24 hours at 47 psig and 271°F, which are the maximum conditions of the design basis accident. This leakage rate is consistent with the construction of the containmnet,[2] which is equipped with a Penetration and Weld Channel Pressurization System for continuously pressurizing both the penetrations and the channels over all containment liner welds. These channels were independently leak-tested during construction.

The safety analysis has been performed on the basis of a leakage rate of 0.10% per day for 24 hours. With this leakage rate and with minimum containment engineered safeguards operating, the public exposure would be well below 10 CFR 100 values in the event of the design basis accident.[3]

The performance of a periodic integrated leakage rate test during plant life provides a current assessment of potential leakage from the containment in case of an accident that would pressurize the interior of the containment. In order to provide a realistic appraisal of the integrity of the containment under accident conditions, this periodic leakage rate test is to be performed without preliminary leak detection surveys or leak repairs, and containment isolation valves are to be closed in the normal manner.

The test pressure of 23.5 psig for the periodic integrated leakage rate test is sufficiently high to provide an accurate measurement of the leakage rate and it duplicates the pre-operational leakage rate test at 23.5 psig. The equations provided relate in a conservative manner the measured leakage of air at 23.5 psig to the potential leakage of a steam-air mixture at 47 psig and 271°F.

The minimum duration of 24 hours for the integrated leakage rate test is established to attain the desired level of accuracy and to allow for daily cyclic variation in temperature and thermal radiation.

The frequency of the periodic integrated leakage rate test is keyed to the schedule for major shutdowns for inservice inspection. The specified frequency of periodic integrated leakage rate is based on the following major considerations.

First is the low probability of leaks in the liner, because of

(a) the tests of the leak tight integrity of the welds during erection;

(b) conformance of the complete containment to a low leakage rate limit at 47 psig during pre-operational testing which is consistent with 0.1% leakage at design basis accident conditions; and

(c) absence of any significant stresses in the liner during reactor operation.

Secondly, the Penetration and Weld Channel Pressurization System is in service continuously to monitor leakage from potential leak paths such as penetrations, liner weld channels, double gasketed seals and spaces between certain containment isolation valves. A leak would be expected to build up slowly and would therefore be noted before design leakage limits are exceeded. Remedial action can be taken before the limit is reached.

The sensitive leakage rate measurements obtained periodically and periodic inspection of accessible portions of the containment wall to detect possible damage to the liner plates, combined with the leakage monitoring afforded by the Penetration and Weld Channel Pressurization System, provide assurance that the containment leakage is within design limits.

The 350 psig test pressure, achieved either by normal Residual Heat Removal System operation or hydrostatically testing, gives an adequate margin over the highest pressure within the system after a design basis accident. Similarly, the hydrostatic test pressure for the containment sump return line of 100 psig gives an adequate margin over the highest pressure within the line after a design basis accident. A recirculation system leakage of 2 gal./hr will limit off-site exposures due to leakage to insignificant levels relative to those calculated for leakage directly from the containment in the design basis accident.

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(1) FSAR - Section 5

(2) FSAR - Section 5.1.7

(3) FSAR - Section 14.3.5

4.5 ENGINEERED SAFETY FEATURES

Applicability

Applies to testing of the Safety Injection System, the Containment Spray System, and the Air Filtration System inside the containment.

Objective

To verify that the subject systems will respond promptly and perform their design functions, if required.

Specification

I. System Tests

A. Safety Injection System

1. System tests shall be performed at each reactor refueling interval. With the Reactor Coolant System pressure less than or equal to 350 psig and temperature less than or equal to 350°F, a test safety injection signal will be applied to initiate operation of the system. The safety injection and residual heat removal pumps are made inoperable for this test.

2. The test will be considered satisfactory if control board indication and visual observations indicate that all components have received the safety injection signal in the proper sequence and timing.

That is, the appropriate pump breakers shall have opened and closed, and all valves shall have completed their travel.

B. Containment Spray System

1. System tests shall be performed at each reactor refueling interval. The tests shall be performed with the isolation valves in the spray supply lines at the containment and the spray additive tank isolation valves blocked closed. Operation of the system is initiated by tripping the normal actuation instrumentation.

2. The spray nozzles shall be checked for proper functioning at least every five years.

3. The test will be considered satisfactory if visual observations indicate all components have operated satisfactorily.

C. Hydrogen Recombiner System

1. A complete recombiner system test shall be performed at each normal reactor refueling on each unit. The test shall include verification of ignition and attainment of normal operating temperature.

2. A complete control system test shall be performed at intervals not greater than six months on each unit. The test shall consist of a complete dry-run startup using artificially generated signals to simulate light off.

3. Containment atmosphere sampling system tests shall be performed at intervals no greater than six months. The test shall include drawing a sample from the fan cooler units and purging the sampling line.

4. The above tests will be considered satisfactory if visual observations and control panel indication indicate that all components have operated satisfactorily.

II. Component Tests

A. Pumps

1. The safety injection pumps, residual heat removal pumps, containment spray pumps and the auxiliary component cooling water pumps shall be started at intervals not greater than one month. The recirculation pumps shall be started during reactor shutdowns for refueling.

2. Acceptable levels of performance shall be that the pumps start, reach their required developed head on recirculation flow, and operate for at least fifteen minutes.

B. Valves

1. Each boron injection tank outlet valve shall be cycled by operator action with the pumps shut down at intervals not greater than once every refueling.

2. Each spray additive valve shall be cycled by operator action with the pumps shut down at intervals not greater than once every refueling.

3. The accumulator check valves shall be checked for operability during each refueling shutdown.

C. Containment Air Filtration System

1. Visual inspection of the filter installation shall be performed every 6 months for the first two years and every refueling thereafter, or at any time work on the filters could alter

their integrity. In addition, measurement of the pressure drop across the moisture separators and HEPA filters shall be performed at each refueling. Any significant difference in appearance or pressure drop from initial conditions shall be corrected. The acceptance criterion is that the pressure drop across the moisture separator and HEPA filters does not exceed 5.0 inches of water at design flow.

2. The iodine removal efficiency of at least one charcoal filter charcoal coupon from each unit shall be measured every six months for the first two years and every refueling thereafter. The charcoal in the coupons shall be from the same batch as that contained in their respective units. The efficiency shall be measured under the containment conditions representative of the design basis accident (47 psig, 271°F, 100% relative humidity and design basis accident iodine concentration and flow). An ignition temperature test shall be performed at the same time. The acceptance criterion for filter efficiency is 50% for removal of methyl iodine and for ignition test that ignition does not occur at 300°C. If the acceptance criteria are not met, an additional coupon from the unit that failed the tests shall be tested. If the second coupon fails to meet the criteria the charcoal contained in that unit shall be replaced.

3. The charcoal filter isolation valves shall be tested at intervals not greater than once every refueling to verify operability.

4. The HEPA filter banks shall be tested with locally generated DOP* at each refueling shutdown and indications of abnormal leakage corrected. The acceptance criterion is that the value of the efficiencies measured during the test shall be at least 99%.

*Dioctylphthalate particles test

D. Hydrogen Recombiner System

1. Each recombiner air-supply blower shall be started at intervals not greater than two months. Acceptable levels of performance shall be that the blowers start, deliver flow, and operate for at least 15 minutes.

Basis:

The Safety Injection System and the Containment Spray System are principal plant safeguards that are normally inoperative during reactor operation. Complete systems tests cannot be performed when the reactor is operating because a safety injection signal causes reactor trip, main feedwater isolation and containment isolation, and a Containment Spray System test requires the system to be temporarily disabled. The method of assuring operability of these systems is therefore to combine systems tests to be performed during annual plant shutdowns, with more frequent component tests, which can be performed during reactor operation.

The annual systems tests demonstrate proper automatic operation of the Safety Injection and Containment Spray Systems. With the pumps blocked from starting a test signal is applied to initiate automatic action and verification made that the components receive the safety injection signal in the proper sequence. The test demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry.[1]

During reactor operation, the instrumentation which is depended on to initiate safety injection and containment spray is generally checked daily and the initiating circuits are tested monthly (in accordance with Specification 4.1). The testing of the analog channel inputs is accomplished in the same manner as for the reactor protection system. The engineered safety features logic system is tested by means of test switches to simulate inputs from the analog channels. The test switches interrupte the logic matrix output to the master relay to prevent actuation. Verification that the logic is accomplished is indicated by the matrix test light. Upon completion of the logic checks, verification that the circuit from the

logic matrices to the master relay is complete is accomplished by use of an ohmmeter to check continuity. In addition, the active components (pumps and valves) are to be tested monthly to check the operation of the starting circuits and to verify that the pumps are in satisfactory running order. The test interval of one month is based on the judgement that more frequent testing would not significantly increase the reliability (i.e., the probability that the component would operate when required), yet more frequent testing would result in increased wear over a long period of time.

Other systems that are also important to the emergency cooling function are the accumulators, the Component Cooling System, the Service Water System and the containment fan coolers. The accumulators are a passive safeguard. In accordance with Specification 4.1 the water volume and pressure in the accumulators are checked periodically. The other systems mentioned operate when the reactor is in operation and by these means are continuously monitored for satisfactory performance.

The charcoal portion of the air recirculation system is a passive safeguard which is isolated from the cooling air flow during normal reactor operation. Hence the charcoal should have a long useful lifetime. The filter frames that house the charcoal are stainless steel and should also last indefinitely. However, the visual inspection specified in Section IIC-1 of this specification will be performed to verify that this is in fact the case. The iodine removal efficiency cannot be measured with the filter cells in place. Therefore at periodic intervals a representative sample of charcoal is to be removed and tested to verify that the efficiencies for removal of methyl iodide, is obtained.[2] The hydrogen recombiner system is an engineered safety feature which will be used only following a loss-of-coolant accident to control the hydrogen evolved in the containment. The system is not expected to be started before about 13 days have elapsed following the accident. At this time the hydrogen concentration in the containment will have reached 2% by volume, which is the design concentration for starting the recombiner system. Actual starting of the system will

be based upon containment atmosphere sample analysis. The complete functional tests of each unit at refueling shutdown will demotrate the proper operation of the recombiner system. More frequent tests of the recombiner control system and air-supply blowers will assure operability of the system. The biannual testing of the containment atmosphere sampling system will demonstrate the availability of this system.

References

(1) FSAR Section 6.2

(2) FSAR Section 6.4

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

4.7 MAIN STEAM STOP VALVES

Applicability

Applies to periodic testing of the main steam stop valves.

Objective

To verify the ability of the main steam stop valves to close upon signal.

Specification

The main steam stop valves shall be tested at refueling intervals with the reactor at cold shutdown. Closure time of five seconds or less shall be verified.

Basis

The main steam stop valves serve to limit an excessive Reactor Coolant System cooldown rate and resultant reactivity insertion following a main steam break incident.[1] Their ability to close upon signal should be verified at each scheduled refueling shutdown. A closure time of five seconds was selected as being consistent with expected response time for instrumentation as detailed in the steam line break incident analysis.[2]

References

(1) FSAR - Section 10.5

(2) FSAR - Section 14.2.5

4.8 AUXILIARY FEEDWATER SYSTEM

Applicability

Applies to periodic testing requirements of the Auxiliary Feedwater System.

Objective

To verify the operability of the Auxiliary Feedwater System and its ability to respond properly when required.

Specification

1.a Each motor driven auxiliary feedwater pump will be started at intervals not greater than every month with full flow established to the steam generators once every refueling.

b The steam turbine driven auxiliary feedwater pump will be started at intervals not greater than six months with full flow established to the steam generators once every refueling.

c The auxiliary feedwater pumps discharge valves will be tested by operator action at intervals not greater than six months.

2. These tests shall be considered satisfactory if control board indication and subsequent visual observation of the equipment demonstrate that all components have operated properly.

Basis

The testing of the auxiliary feedwater pumps will verify their operability. The capacity of any one of the three auxiliary feedwater pumps is sufficient to meet decay heat removal requirements.

Verification of correct operation will be made both from instrumentation within the main control room and direct visual observation of the pumps.

Reference

FSAR - Sections 10.4, 14.1.9 and 14.2.5

4.9 REACTIVITY ANOMALIES

Applicability

Applies to potential reactivity anomalies.

Objective

To require evaluation of reactivity anomalies within the reactor.

Specification

Following a normalization of the computed boron concentration as a function of burn-up, the actual boron concentration of the coolant shall be periodically compared with the predicted value. If the difference between the observed and predicted steady-state concentrations reaches the equivalent of one percent in reactivity, the Atomic Energy Commission shall be notified within 24 hours and an evaluation as to the cause of the discrepancy shall be made and reported to the Atomic Energy Commission within 10 days.

Basis

To eliminate possible errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burn-up and the boron concentration, necessary to maintain adequate control characteristics, must be adjusted (normalized) to accurately reflect actual core conditions. When full power is reached initially, and with the control rod groups in the desired positions, the boron concentration is measured and the predicted curve is adjusted to this point. As power operation proceeds, the measured boron concentration is compared with the predicted concentration and the slope of the curve relating burn-up and reactivity is compared with that predicted. This process of normalization shall be completed early in core life. Thereafter, actual boron concentration can be compared with prediction, and the reactivity status of the core can be continuously evaluated. Any reactivity anomaly greater than 1% would be unexpected, and its occurrence would be thoroughly investigated and evaluated. The value of 1% is considered a safe limit since a shutdown margin of at least 1% with the most reactive rod in the fully withdrawn position is always maintained.

4.10 ENVIRONMENTAL MONITORING SURVEY

Applicability

Applies to routine testing of the plant environs.

Objective

To establish a sampling schedule which will recognize changes in radioactivity in the environs, and assure that effluent releases are kept as low as practicable and within allowable limits.

Specification

1. Liquid Discharges

   The survey for liquid discharges shall be conducted in accordance with Table 4.10-1 as specified below:

   a. If the gross beta-gamma activity of the station releases to the river is less than 1% of MPC during the month just ended, the environmental survey shall be conducted in accordance with Program 1 for the subsequent month.

   b. If the gross beta-gamma activity of the station releases to the river is greater than 1% of MPC but less than 10% of MPC during the month just ended, the environmental survey shall be conducted in accordance with Program 2 for the subsequent month. If the samples taken under Program 2 do not indicate any significant increase in environmental radioactivity, the survey shall revert to Program 1.

   c. If the gross beta-gamma activity of the station releases to the river is greater than 10% of MPC during the month just ended, the environmental survey shall be conducted in accordance

with Program 3 for the subsequent month. If the samples taken under Program 3 do not indicate any significant increase in environmental radioactivity, the survey shall revert to Program 2.

d. Irrespective of release levels, once each year the survey shall be taken under Program 3 for a 3 month continuous period.

2. Gaseous Discharges

The survey for the gaseous discharges shall be conducted in accordance with Table 4.10-2 as specified below:

a. If the average release rate from the plant vent is less than 1% of the annual allowable release rate as specified in Paragraph 3.9-C1 during the month just ended, the environmental survey shall be conducted in accordance with Program 1 for the subsequent month.

b. If the average release rate from the plant vent is greater than 1% but less than 10% of the annual allowable release rate as specified in Paragraph 3.9-C1 during the month just ended, the environmental survey shall be conducted in accordance with Program 2 for the subsequent month. If the samples taken under Program 2 do not indicate any significant increase in environmental radioactivity, the survey shall revert to Program 1.

c. If the average release rate from the plant vent is greater than 10% of the annual allowable release rate as specified in Paragraph 3.9-C1 during the month just ended, the environmental survey shall be conducted in accordance with Program 3 for the subsequent month. If the samples taken under Program 3 do not indicate any significant increase in environmental radioactivity, the survey shall revert to Program 2.

d. Irrespective of release levels, once each year the survey shall be taken under Program 3 for a 3 month continuous period.

Basis

Programs for monitoring the adjacent area of the Hudson River will be conducted by the Consolidated Edison Company, by the New York State Department of Health, and by the New York University Institute of Environmental Medicine. The New York State program includes measurement of samples of air, water, milk and wildlife. The New York University Medical Center research program includes the biology of the Hudson River, the distribution and abundance of fish in the river, pesticides and radio-ecological studies.

A nineteen month study which began in June, 1969 is being conducted by Raytheon for the Hudson River Policy Committee. The Committee consists of the New York State Conservation Department, the New Jersey Department of Conservation and Economic Development, the U. S. Bureau of Sport Fisheries and Wildlife, the U. S. Bureau of Commercial Fisheries, and the Connecticut Conservation Department. The objectives of the study are; (1) to determine the seasonal distribution of fish and key organisms within and outside of the areas to be exposed to the heated and otherwise altered discharge from Units 1, 2, and 3; (2) to determine the effects of temperature rise and chemical additives on the survival and behavior of screenable and non-screenable fish and organisms in the area; (3) to catalog physical and chemical characteristics of the estuary often associated with observed changes in the biota; i.e., temperature, salinity, conductivity, dissolved and suspended solids, dissolved oxygen, and physical alternations.

The various studies mentioned above include measurements of radioactivity in fresh water, river water, river sediments, fish, milk, aquatic vegetation, vegetation, soil, and air in the vicinity of the Indian Point Station.

The environmental monitoring program conducted by the Consolidated Edison Company will supply sufficient data to determine the compliance of the Indian Point Station with the requirements of 10CFR20. The schedules for liquid and gaseous discharges will insure that changes in the environmental radioactivity will be detected.

Although the design of the proposed facility and administrative controls will be such that gaseous and liquid effluents will be released in accordance with the requirements of 10CFR20, the environmental monitoring program of the Consolidated Edison company provides a redundant means of insuring that the operation of the proposed facility does not pose any undue risk to the health and safety of the public. The New York State and New York University programs provide an independent means of verifying the proposed facilities compliance with 10CFR20.

Table 4.10-1

Environmental Monitoring Survey - Liquid Discharges+

Media of Sample	No. of Samples/ Collection	Programs 1 Collection Frequency	Programs 1 Analysis*	Programs 2 Collection Frequency	Programs 2 Analysis*	Programs 3 Collection Frequency	Programs 3 Analysis
Hudson River Water	2 1	W MC	GBG T	TW MC	GBG GSA T	D MC	GBG GSA RA T
Hudson River Aquatic Vegetation	15	SSF	GBG	MDGS	GBG GSA	MDGS	GBG GSA RA
Hudson River Bottom Sediment	5	SSF	GBG	M	GBG GSA	M	GBG GSA RA
Hudson River Fish	1	M	GBG	TM	GBG GSA	W	GBG GSA RA

+Samples will be taken whenever biologically available.

*Minimum equipment sensitivity shall be those given in FSAR Table 11.11-1.

Nomenclature for Sample Frequency

W - Weekly
TW - Twice Weekly
D - Daily
M - Monthly
MC - Monthly Composite
TM - Twice Monthly
SSF - Once each in Spring, Summer and Fall
MDGS - Monthly During the Growing Season

Nomenclature for Analysis

GBG - Gross Beta-Gamma GSA - Gamma Spectrometer Analysis T - Tritium
RA - Radiochemical Analysis to determine biologically important isotopes.

Table 4.10-2

Environmental Monitoring Survey - Gaseous Discharges+

		PROGRAMS					
		1		2		3	
Media of Sample	No. of Samples/ Collection	Collection Frequency	Analysis*	Collection Frequency	Analysis**	Collection Frequency	Analysis
Fallout	2	M	GBG T*	M	GBG GSA T*	TM MC	GBG GSA RA T*
Air Particulate & Organic Iodide	5	W	GBG GSA	TW	GBG GSA	TW	GBG GSA RA
Drinking Water Supplies	3	M	GBG T	TM MC	GBG GSA T	W MC	GBG GSA RA T
Lake Water & Well Water	6	M	GBG T	TM MC	GBG GSA T	W MC	GBG GSA RA T
Lake Aquatic & Vegetation & Land Vegetation	8	SSF	GBG	MDGS	GBG GSA	MDGS	GBG GSA RA
Soil	5	A	GBG	M	GBG GSA	M	GBG GSA RA
Direct Gamma (Spot Readings)	180	A	GGB	MSL	GGB	WSL	GBG

Table 4.10-2 (Continued)

		PROGRAMS					
		1		2		3	
Media of Sample	No. of Samples/ Collection	Collection Frequency	Analysis*	Collection Frequency	Analysis**	Collection Frequency	Analysis
Direct Gamma (Peripheral Monitoring)	15	M	GGB	TM	GGB	W	GGB
Milk	1					M	GBG GSA RA

+Samples will be taken whenever biologically available.

*Tritium analysis will be performed provided sufficient wet deposition occurs.

**Minimum equipment sensitivities shall be those given in FSAR Table 11.11-1

Table 4.10-2 (Continued)

Environmental Monitoring Survey - Gaseous Discharge

Nomenclature for Sample Frequency

M - Monthly
TM - Twice Monthly
W - Weekly
TW - Twice Weekly
MC - Monthly Composite
A - Annually
SSF - Once each in Spring, Summer and Fall
MDGS - Monthly During the Growing Season
MSL - Monthly at Selected Locations
WSL - Weekly at Selected Locations

Nomenclature for Analysis

GBG - Gross Beta-Gamma
GSA - Gamma Spectrometer Analysis
RA - Radiochemical Analysis to determine biologically important isotopes
T - Tritium
GGB - Gross Gamma Background

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 recireculation 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. All of the fan cooler units are equipped with activated charcoal filters to remove volatile iodine following an 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 2.8 weight per cent of U-235. Three fuel enrichments are used in the initial core. The highest enrichment is a nominal 3.3 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 1412 poison rods in the form of 7,8,9,12,16 and 20-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 & Sec. 3 of Fuel Densification Indian Point Nuclear Generating Station Unit No. 2, Dated January, 1973

(2) FSAR Section 3.2.1 & Sec. 3 of Fuel Densification Indian Point Nuclear Generating Station Unit No. 2, Dated January, 1973

(3) FSAR Section 3.2.1 & Figure 3.3 of Fuel Densification Indian Point Nuclear Generating Station Unit No. 2, Dated January, 1973

(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 Station Manager.

2. The 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.

3. The normal organization for conduct of operation of the Nuclear Power Generation Department, and specifically, the Unit No. 2 facility, is shown on Figure 6.1-1.

   Until such time as the Rector Operator for Unit No. 2 obtains a Reactor Operator's license for Unit No. 2 to meet the shift staffing requirement, the shift complement will be amended such that there will be:

(1) An Engineer or other person holding equal or better qualifications of those Production Engineers described in the Final Safety Analysis Report holding a Senior Reactor Operator's license on Unit No. 2 serving as Technical Advisor on Unit No. 2, or

(2) A PWR experienced reactor operator serving as a second control operator on Unit No. 2 with an individual holding at least the qualifications of (1) above being readily available on call, or

(3) A representative of the plant vendor who by virtue of his training and experience in PWR operation can provide competent on-shift technical support.

4. During cold shutdown conditions, when fuel is in the reactor, the minimum functional operating organization for the Unit No. 2 facility shall include:

a) One individual licensed on Unit No. 2 at the Reactor Operator level pursuant to 10CFR55. This operator shall be in the control room at all times and,

b) One Individual licensed on Unit No. 2 at the Senior Operator level pursuant to 10CFR55, and,

c) A Health Physics Technician on-site at all times when nuclear fuel is located therein. This technician shall assume parallel duties on Unit No. 1.

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.

The Chairman may establish subcommittees and designate members of the full committee for assignment to the subcommittees. The Chairman, Vice Chairman and Secretary of the Nuclear Facilities Safety Committee are ex-officio members of all subcommittees.

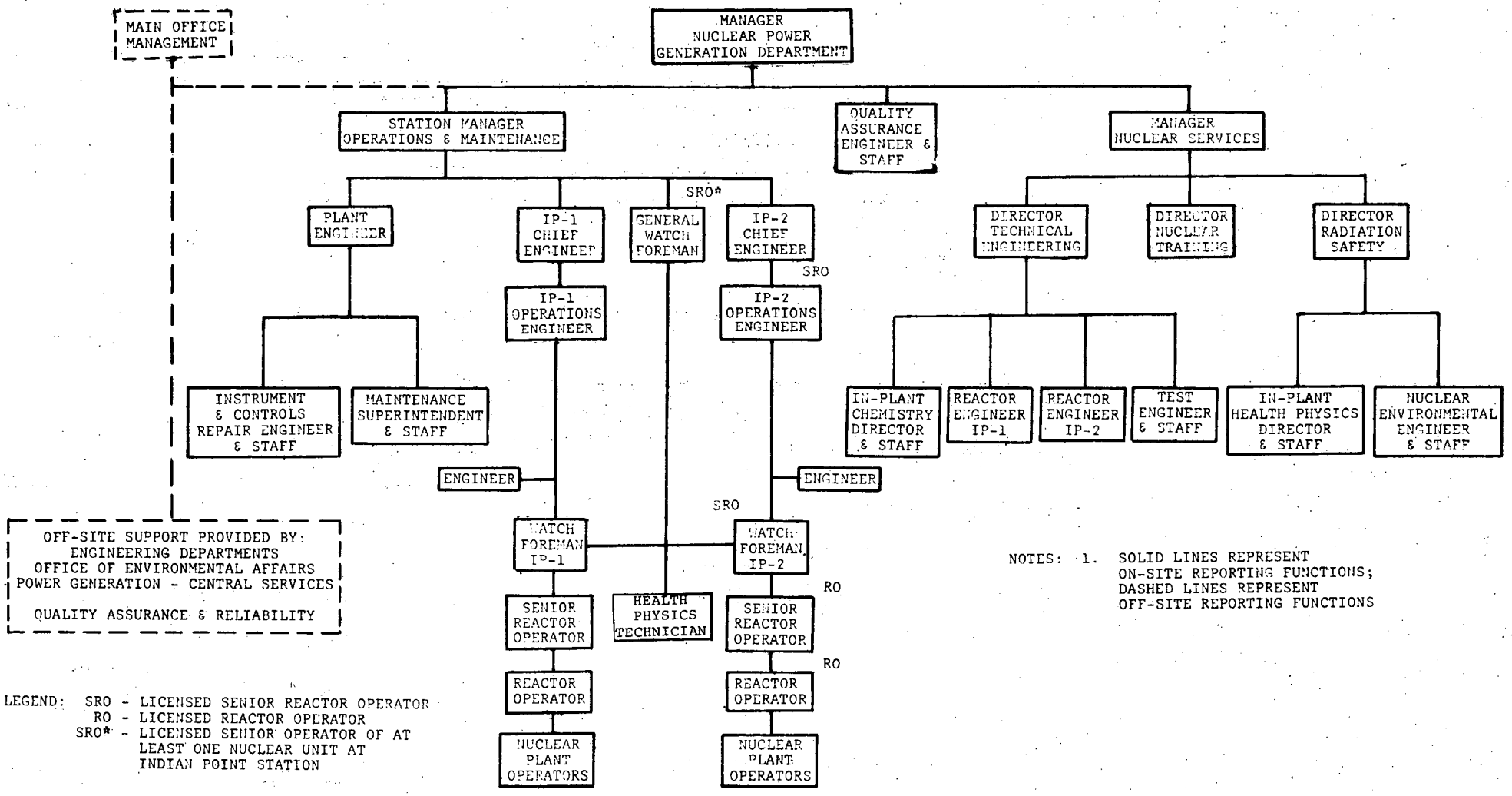
f. Records

Draft minutes of all meetings will be distributed promptly to each committee member for comment and any corrections. Amended and corrected minutes will be circulated to committee members for final approval. Copies of approved minutes will be promptly distributed to each committee member and to the Assistant Vice President, Power Generation, the Vice President, Power Supply, the Executive Vice President, Central Operations, the President, the Chairman of the Board, and the Corporation Secretary.

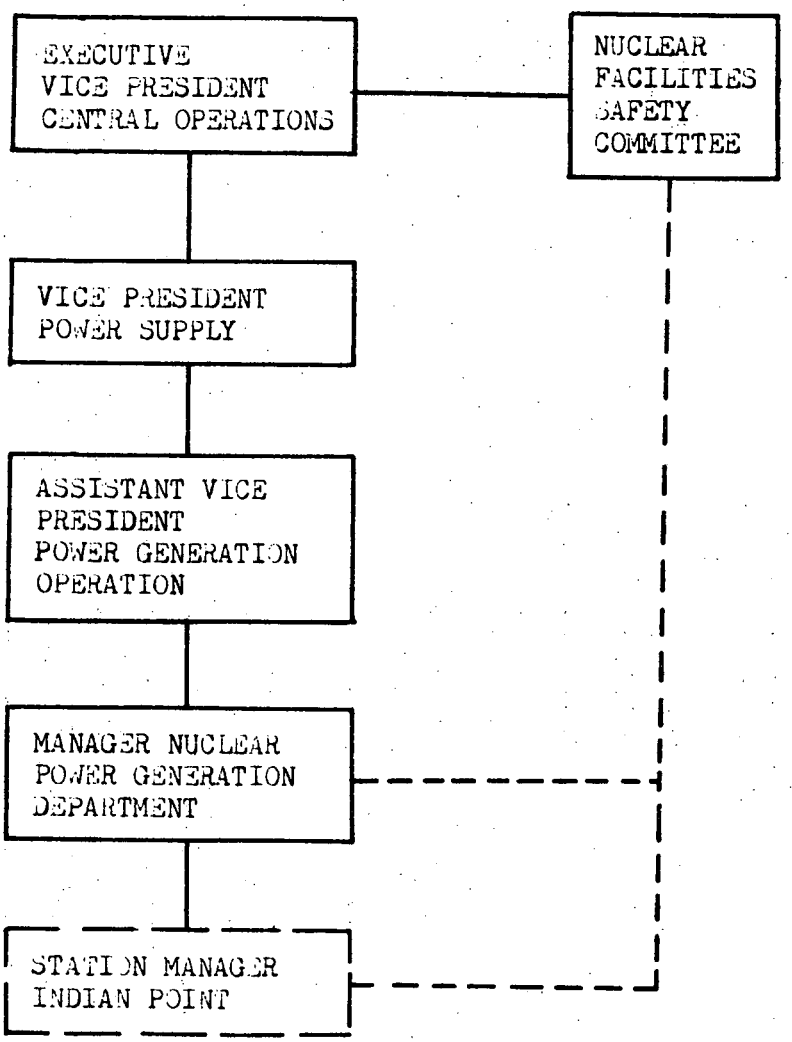
Special reports shall be in writing and will be distributed to all members of the Committee, the Assistant Vice President, Power Generation, the Vice President, Power Supply, the Executive Vice President, Central Operations, the President, the Chairman of the Board, and the Corporation Secretary.

g. Charter

A written charter delineating the establishment, composition, mission, and the dissemination of minutes and reports shall be maintained which may be amended as required.



STATION ORGANIZATION CHART
FIGURE 6.1-1



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 normal plant operation, including power levels and periods of operation at each power level.

b. Records of principal maintenance activities.

c. Records of plant shutdowns.

d. Records of abnormal occurrences.

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

f. Records of reactor tests and measurements.

g. 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.

c. Records of plant radiation and contamination surveys.

d. Records of off-site environmental monitoring surveys.

e. Records of radiation exposure for all station personnel, contractors and visitors to the plant who enter the controlled area.

f. Records of radioactive waste disposal.

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. <u>Operations Summary</u>

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.

B. Power Generation

A summary of the nuclear and electrical output of the plant during the reporting period, and the cumulative total outputs since initial criticality, including:

1. Gross thermal power generated (in MWH).
2. Gross electrical power generated (in MWH).
3. Net electrical power generated (in MWH).
4. Number of hours the reactor was critical.
5. Number of hours the generator was on line.
6. Histogram of thermal power versus time.

C. Shutdowns

Descriptive material covering all outages occurring during the reporting period. The following information shall be provided for each outage:

1. The cause of the outage.
2. The method of shutting down the reactor; e.g., scram, automatic rundown, or manually controlled deliberate shutdown.
3. Duration of the outage in hours.
4. Plant status during the outage; e.g., cold shutdown or hot standby.
5. Corrective and preventive action taken to preclude recurrence of each unplanned outage.

D. Maintenance

A discussion of electrical, mechanical and general maintenance performed during the report period having potential effects on the safety of the facility.

Specific systems involved shall be identified and information shall be provided on:

1. The nature of the maintenance; e.g., routine or emergency.
2. The effect, if any, on the safe operation of the reactor.
3. The cause of any malfunction for which corrective maintenance was required.
4. The results of any such malfunction.
5. Corrective and preventive action taken to preclude recurrence.

E. Radioactive Effluent Releases

Information relative to the quantities of liquid, gaseous and solid radioactive effluents released from the facility, and the effluent volumes used in maintaining the releases within the limits of 10 CFR 20 shall be provided as follows:

1. Liquid Releases

   a. Total radioactivity (in curies) released, other than tritium, and average concentration at outfall of discharge canal.

   b. Total tritium oxide (in curies) discharged, and average concentration at outfall of discharge canal.

   c. Total volume (in gallons) of liquid waste released into discharge canal.

   d. Total volume (in gallons) of dilution water used.

   e. The maximum concentration released (averaged over the period of a single release).

f. Estimated total radioactivity (in curies) released, by nuclide (other than tritium), based on representative isotopic analyses performed.

g. Percent of applicable limits released, based on nuclide identification performed.

2. Gaseous Releases

a. Total radioactivity (in curies) released of:

(1) noble gases
(2) halogens
(3) particulates with half-lives greater than eight days
(4) tritium oxide

b. Maximum release rate (for any one-hour period).

c. Estimated total radioactivity (in curies) released, by nuclide (other than tritium), based on representative isotopic analyses performed.

d. Percent of applicable limit released, based on nuclide identification performed.

3. Solid Waste

a. The total amount of solid waste packaged (in cubic feet).

b. Estimated total radioactivity (in curies) involved.

c. The dates of shipment and disposition (if shipped off-site).

F. Environmental Monitoring

1. For each medium sampled during the six-month period, the following information shall be provided:

   a. Number of sampling locations

   b. Total number of samples

   c. Number of locations at which levels are found to be significantly greater than local backgrounds.

   d. Highest, lowest, and the annual average concentrations or levels of radiation for the sampling point with the highest average and description of the location of that point with respect to the site.

2. If levels of plant contributed radioactive materials in environmental media indicate the likelihood of public intakes in excess of 3% of those that could result from continuous exposure to the concentration values listed in Appendix B, Table II, Part 20, estimates of the likely resultant exposure to individuals and to population groups, and assumptions upon which estimates are based shall be provided. (These values are comparable to the top of Range I, as defined in FRC Report No. 2.)

3. If statistically significant variations in off-site environmental concentrations with time are observed, and are attributed to plant releases correlation of these results with effluent releases shall be provide.

6.6.5 Special Reports:

A. A report* within 60 days of completion of one year of commercial operation covering:

1. An evaluation of plant performance to date in comparison with design specifications.

2. A reassessment of the validity of prior accident analyses in light of measured operating characteristics, which may affect consequences; and system, component and personnel performance which may affect accident probabilities.

3. A progress and status report on all items identified in the operating license review as requiring further effort.

*This report may be incorporated in the Semi-Annual Operations Report due at that time.

B. Each containment integrated leak rate test shall be the subject of a summary technical report including results of the local leak rate tests since the last report. The report shall be submitted within 90 days to the Director, Divison of Reactor Licensing, USAEC, Washington, D.C. 20545 and shall include analyses and interpretations of the results which demonstrate compliance in meeting the leak rate limits specified in the Technical Specifications.

C. A report covering the X-Y xenon stability tests within three months upon completion of the tests.

D. To provide the Commission with added verifications of the safety and reliability of the pre-pressurized Zircaloy-clad nuclear fuel, a limited program of non-destructive fuel inspections will be conducted at Indian Point Unit 2. The program shall consist of a visual inspection (e.g., underwater TV, periscope, or other) of the two lead burnup assemblies in each region during the first, second, and third refueling shutdowns. Any condition observed by this inspection which would lead to unacceptable fuel performance may be the object of an expanded surveillance effort. If another domestic plant which contains pre-pressurized fuel of a similar design reaches fuel exposures equal to or greater than at Indian Point Unit 2, and if a limited inspection program is or has been performed there, then the program may not have to be performed at Indian Point Unit 2. However, such action requires approval of the Atomic Energy Commission. The results of these inspections will be reported to the Atomic Energy Commission.

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.