APPENDIX A

TO

FACILITY OPERATING LICENSE DPR-64

FOR

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

DOCKET NO. 50-286

TECHNICAL SPECIFICATIONS

FOR

FUEL LOADING AND SUBCRITICAL TESTING

Date of Issuance: DEC 12 1975

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1.0 DEFINITIONS

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

1.1 REACTOR CONDITIONS

1.1.1 Rated Power

A steady state reactor core output of 3025 MWt.

1.1.2 Reactor Pressure

The pressure in the steam space of the pressurizer.

1.1.3 T_{avg}

Average temperature across the reactor vessel as measured by the hot and cold leg temperature detectors.

1.2 REACTOR OPERATING CONDITIONS

1.2.1 Cold Shutdown Condition

When the reactor is subcritical by at least 1% $\Delta k/k$ and T_{avg} is $\leq 200°F$.

1.2.2 Hot Shutdown Condition

When the reactor is subcritical, by an amount greater than or equal to the margin as specified in Specification A-1 and T_{avg} is $>200°F$ but $\leq 555°F$.

1.2.3 Reactor Critical

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

1.2.4 Refueling Operation Condition

When the reactor is subcritical by at least 10% $\Delta k/k$ and T_{avg} is $\leq 140°F$ and core alterations are being made with the head completely unbolted.

1.3 CORE ALTERATION

The addition, removal, relocation or other movement of fuel, controls, or installed equipment or material in a reactor core, except for functions normally performed during conventional reactor operation in accordance with intended design of equipment, such as control rod or instrument detector movement or performance of flux scans.

1.4 OPERABLE

Properly installed in the system and capable of performing the intended functions in the intended manner as verified by testing, and tested at the frequency required by the Technical Specifications.

1.5 OPERATING/INSERVICE

Performing the intended functions in the intended manner.

1.6 PROTECTION INSTRUMENTATION AND LOGIC

The protection system consists of the actuation devices of both the reactor protection system and the engineered safety features systems.

1.6.1 Instrument Channel

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

1.6.2 Logic Channel

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

1.7 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.

1.8 INSTRUMENTATION SURVEILLANCE

1.8.1 Instrument Channel Check

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

1.8.2 Instrument Channel Functional Test

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

1.8.3 Instrument 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.

1.8.4 Logic Channel Functional Test

The operation of relays or switch contacts, in all the combinations required, to produce the required output.

1.9 CONTAINMENT INTEGRITY

Containment integrity is defined to exist when:

1.9.1 All non-automatic containment isolation valves which are not required to be open during accident conditions, except those required to be open for normal plant operation or testing as identified in Table 3.6-1, are closed and blind flanges are installed where required.

1.9.2 The equipment door is properly closed.

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

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

1.9.5 The containment leakage satisfies Specification 4.4.

1.10 LOCKED VALVES

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

1.11 ABNORMAL OCCURRENCE

An abnormal occurrence shall be any of those conditions specified below:

1.11.1 Events requiring prompt notification of the NRC in accordance with Section C.2a of Revision 3 of Regulatory Guide 1.16 are as follows:

a. Failure of the reactor protection system or other systems subject to limiting safety-system settings to initiate the required protective function by the time a monitored parameter reaches the setpoint specified as the limiting safety system setting in the technical specifications or failure to complete the required protective function.

b. Operation of the unit or affected systems when any parameter or operation subject to a limiting condition for operation is less conservative than the least conservative aspect of the limiting condition for operation established in the technical specifications.

c. Abnormal degradation discovered in fuel cladding, reactor coolant pressure boundary, or primary containment.

d. Reactivity anomalies involving disagreement with the predicted value of reactivity balance under steady-state conditions during power operation greater than or equal to 1% $\Delta k/k$; a calculated reactivity balance indicating a shutdown margin less conservative than specified in the technical specifications; short-term reactivity increases that correspond to a reactor period of less than 5 seconds or, if subcritical, an unplanned reactivity insertion of more than 0.5% $\Delta k/k$; or occurrence of any unplanned criticality.

e. Failure or malfunction of one or more components which prevents or could prevent, by itself, the fulfillment of the functional requirements of system(s) used to cope with accidents analyzed in the FSAR.

f. Personnel error or procedural inadequacy which prevents or could prevent, by itself, the fulfillment of the functional requirements of systems required to cope with accidents analyzed in the FSAR.

g. Conditions arising from natural or man-made events that, as a direct result of the event, require plant shutdown, operation of safety systems, or other protective measures required by technical specifications.

h. Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the FSAR or in the bases for the technical specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

i. Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than that assumed in the accident analyses in the FSAR or technical specifications bases; or discovery during plant life of conditions not specifically considered in the FSAR or technical specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

1.11.2 Events requiring the submittal of a written report to the NRC within 30 days of occurrence in accordance with Section C.2b of Revision 3 of Regulatory Guide 1.16 are as follows:

a. Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the technical specifications, but which do not prevent the fulfillment of the functional requirements of affected systems.

b. Conditions leading to operation in a degraded mode permitted by a limiting condition for operation, or plant shutdown required by a limiting condition for operation.

c. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety features.

d. Abnormal degradation of systems other than those specified in 1.11.1c, above, designed to contain radioactive material resulting from the fission process.

1.12 SURVEILLANCE INTERVAL

When Refueling Outage is used to designate a surveillance interval, the surveillance will be performed during the refueling outage or up to six months before the refueling outage. When a refueling outage occurs within 8 months of the previous refueling outage, the surveillance testing need not be performed. The maximum interval between surveillance tests is 18 months.

Surveillance intervals, with the exception of refueling, shift and daily periods, are defined as the specified period plus or minus 25% of the specified period.

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, the system is hydrotested in accordance with Power Piping Code USAS B31.1 (1967) prior to initial operation.

<u>References</u>

(1) FSAR Section 4

(2) FSAR Table 4.1-1

(3) FSAR Section 4.3.4

3 LIMITING CONDITIONS FOR OPERATION

For the cases where no exception time is specified for inoperable components, this time is assumed to be zero.

3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the Reactor Coolant System; operational components; heatup, cooldown, criticality, activity, chemistry and leakage.

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.

2. 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 Section XI of the ASME Boiler and Pressure Vessel Code.

b. All pressurizer code safety valves shall be operable whenever the reactor is above the cold shutdown condition except during reactor coolant system hydrostatic tests and/or safety valve settings.

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.

Each of the pressurizer code safety valves is designed to relieve 420,000 lbs. per hr. of saturated steam at the valve setpoint.

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 overpressurization.

3.1.B HEATUP AND COOLDOWN

Specifications

1. The reactor coolant temperature and pressure and system heatup and cooldown rates averaged over 1 hour (with the exception of the pressurizer) shall be limited in accordance with Figures 3.1-1 and 3.1-2 for the first full-power service period.

   a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.

   b. Figures 3.1-1 and 3.1-2 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. The limit lines shown in Figures 3.1-1 and 3.1-2 shall be recalculated periodically using methods discussed in the Basis and the results of surveillance specimens.

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

4. The pressurizer heatup and cooldown rates averaged over 1 hour shall not exceed 200°F/hr. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

5. Reactor Coolant System Hydrostatic Tests shall be performed in accordance with Section 4.3.

Basis

Fracture Toughness Properties

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the Summer 1965 Section III of the ASME Boiler and Pressure Vessel Code Reference (6) and ASTM E185 Reference (5) and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1972 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel Code, Reference (1), and the calculation methods described in Reference (2).

Heatup and cooldown limit curves are calculated using the most limiting value of RT_{NDT} at the end of 2 years of service life. The 2-year service life period is chosen such that the limiting RT_{NDT} at the 1/4 T location in the core region is higher than the RT_{NDT} of the limiting unirradiated material. This service period assures that all components in the Reactor Coolant System will be operated conservatively in accordance with Code recommendations.

The highest RT_{NDT} of the core region material is determined by adding the radiation induced ΔRT_{NDT} for the applicable time period to the original RT_{NDT} shown in Table Q4.2-1[3]. The fast neutron (E >1 MEv) fluence at 1/4 thickness and 3/4 thickness vessel locations is given as a function of full-power service life in revised Figure 4.2-10[4]. Using the applicable fluence at the end of the 2-year period and the copper content of the material in question, the ΔRT_{NDT} is obtained from Figure 4.4-3[4].

Values of ΔRT_{NDT} determined in this manner may be used until the results from the material surveillance program, when evaluated according to ASTM E185, are available. The first capsule will be removed early in the service life of the reactor vessel, note FSAR Section 4.5.1. The heatup and cooldown curves will be re-evaluated if the ΔRT_{NDT} determined from the surveillance capsule is different from the predicted ΔRT_{NDT}.

Heatup and Cooldown Curves

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Non-mandatory Appendix G2000 in Section III of the ASME Boiler and Pressure Vessel Code; and discussed in detail in Reference (2).

The approach specifies that the allowable total stress intensity factor (K_I) at any time during heatup or cooldown cannot be greater than that shown on the K_{IR} curve (Reference 1) for the metal temperature at that time.

Furthermore, the approach applies explicit safety factor of 2.0 on the stress intensity factor induced by pressure gradients. Thus, the governing equation for the heatup - cooldown analysis is:

$$2\ K_{Im} + 1.0\ K_{It} \leq K_{IR} \qquad (1)$$

where:

K_{Im} is the stress intensity factor caused by membrane (pressure) stress

K_{It} is the stress intensity factor caused by the thermal gradients

K_{IR} is provided by the code as a function of temperature relative to the RT_{NDT} of the material.

During the heatup analysis, Equation (1) is evaluated for two distinct situations.

First, allowable pressure-temperature relationships are developed for steady-state (i.e., zero rate of change of temperature) conditions assuming the presence of the code reference 1/4 T deep flaw at the ID of the pressure vessel. Due to the fact that, during heatup, the thermal gradients in the vessel wall tend to produce compressive stresses at the 1/4 T location, the tensile stresses induced by internal pressure are somewhat alleviated. Thus, a pressure-temperature curve based on steady-state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the 1/4 T location is treated as the governing factor.

The second portion of the heatup analysis concerns the calculation of pressure temperature limitations for the case in which the 3/4 T location becomes the controlling factor. Unlike the situation at the 1/4 T location, at the 3/4 T position (i.e., the tip of the 1/4 T deep O.D. flaw) the thermal gradients established during heatup produce stresses which are tensile in nature; and thus, tend to reinforce the pressure stresses present. These thermal stresses are, of course, dependent on both the rate of heatup and the time (or water temperature) along the heatup ramp. Furthermore, since the thermal stresses at 3/4 T are tensile and increase with increasing heatup rate, a lower bound curve similar to that described in the preceding paragraph cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced in the following fashion. First, a composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the two values taken from the curves under consideration. The composite curve is then adjusted to allow for possible errors in the pressure and temperature sensing instruments.

The use of the composite curve becomes mandatory in setting heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp, the controlling analysis switches from the O.D. to the I.D. location; and the pressure limit must, at all times, be based on the most conservative case.

The cooldown analysis proceeds in the same fashion as that for heatup, with the exception that the controlling location is always the 1/4 T. The thermal gradients induced during cooldown tend to produce tensile stresses at the 1/4 T location and compressive stresses at the 3/4 T position. Thus, the ID flaw is clearly the worst case.

As in the case of heatup, allowable pressure temperature relations are generated for both steady-state and finite cooldown rate situations. Composite limit curves are then constructed for each cooldown rate of interest. Again, adjustments are made to account for pressure and temperature instrumentation error.

The use of the composite curve in the cooldown analysis is necessary because system control is based on a measurement of reactor coolant temperature, whereas the limiting pressure is calculated using the material temperature at the tip of the assumed reference flaw. During cooldown, the 1/4 T vessel location is at a higher temperature than the fluid adjacent to the vessel I.D. This condition is, of course, not true for the steady-state situation. It follows that the ΔT induced during cooldown results in a calculated higher allowable K_{IR} for finite cooldown rates than for steady-state under certain conditions.

Because operation control is on coolant temperature, and cooldown rate may vary during the cooldown transient, the limit curves shown in Figure 3.1-2 represent a composite curve consisting of the more conservative values calculated for steady-state and the specific cooling rate shown.

Details of these calculations are provided in Reference (2).

Pressurizer Limits

Although the pressurizer operates at temperature ranges above those for which there is reason for concern about brittle fracture, operating limits are provided to assure compatability of operation with the fatigue analysis performed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, 1965 Edition and Associated Code Addenda through the Summer 1966 Addendum.

<u>References</u>

(1) ASME Boiler and Pressure Vessel Code, Section III, 1972 Summer Addenda

(2) WCAP-7924, "Basis for Heatup and Cooldown Limit Curves", W. S. Hazelton, S. L. Anderson, S. E. Yanichko, July 1972

(3) FSAR Volume 5, Response to Question Q4.2.

(4) FSAR Section 4.

(5) ASTM E185-70, Surveillance Tests on Structural Materials in Nuclear Reactors.

(6) ASME Boiler and Pressure Vessel Code, Section III, Summer 1965.

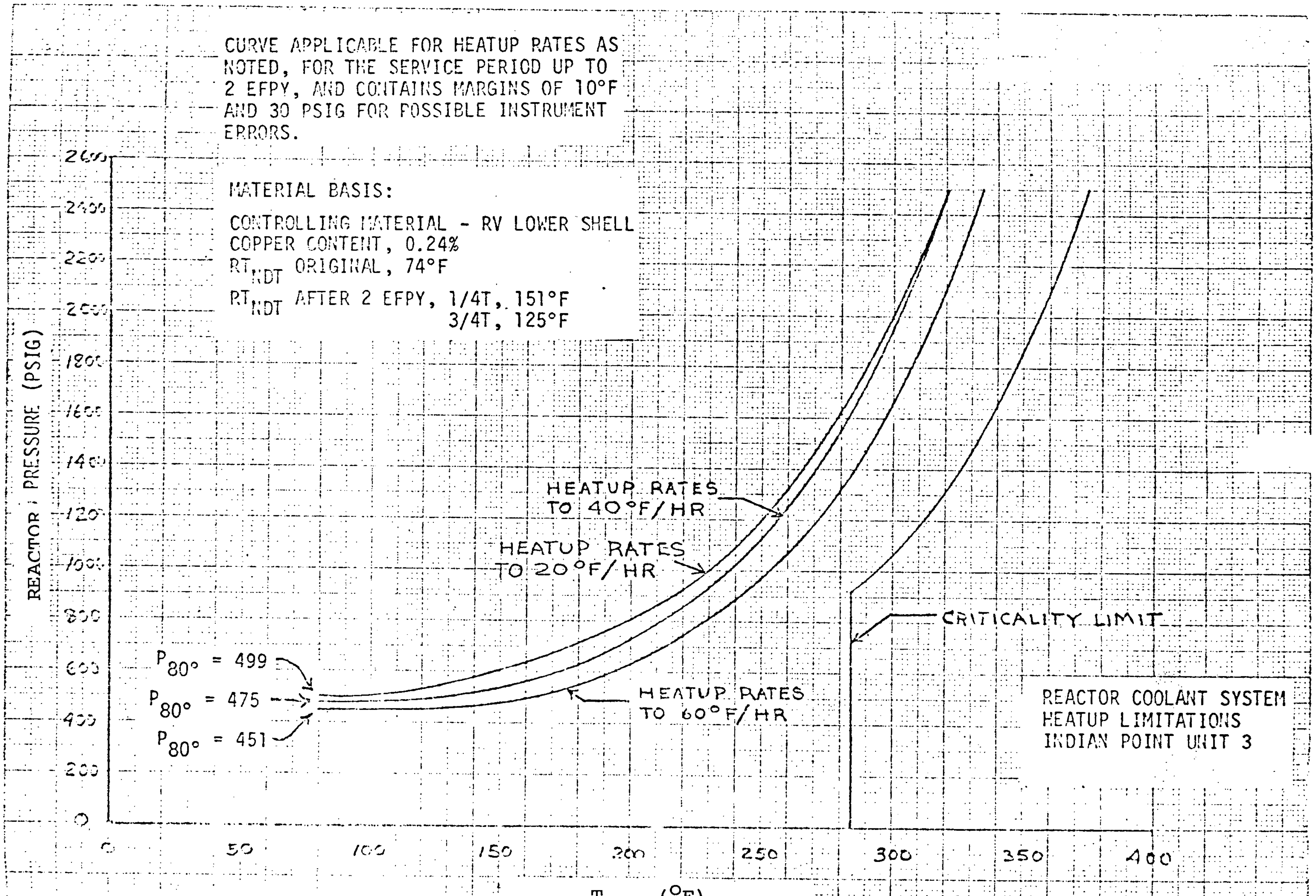
CURVE APPLICABLE FOR HEATUP RATES AS NOTED, FOR THE SERVICE PERIOD UP TO 2 EFPY, AND CONTAINS MARGINS OF 10°F AND 30 PSIG FOR POSSIBLE INSTRUMENT ERRORS.
MATERIAL BASIS:
CONTROLLING MATERIAL - RV LOWER SHELL
COPPER CONTENT, 0.24%
RT_{NDT} ORIGINAL, 74°F
RT_{NDT} AFTER 2 EFPY, 1/4T, 151°F
3/4T, 125°F
HEATUP RATES TO 40°F/HR
HEATUP RATES TO 20°F/HR
HEATUP RATES TO 60°F/HR
CRITICALITY LIMIT
$P_{80°}$ = 499
$P_{80°}$ = 475
$P_{80°}$ = 451
REACTOR COOLANT SYSTEM HEATUP LIMITATIONS INDIAN POINT UNIT 3
REACTOR PRESSURE (PSIG)
T_{AVG} (°F)

Figure 3.1-1

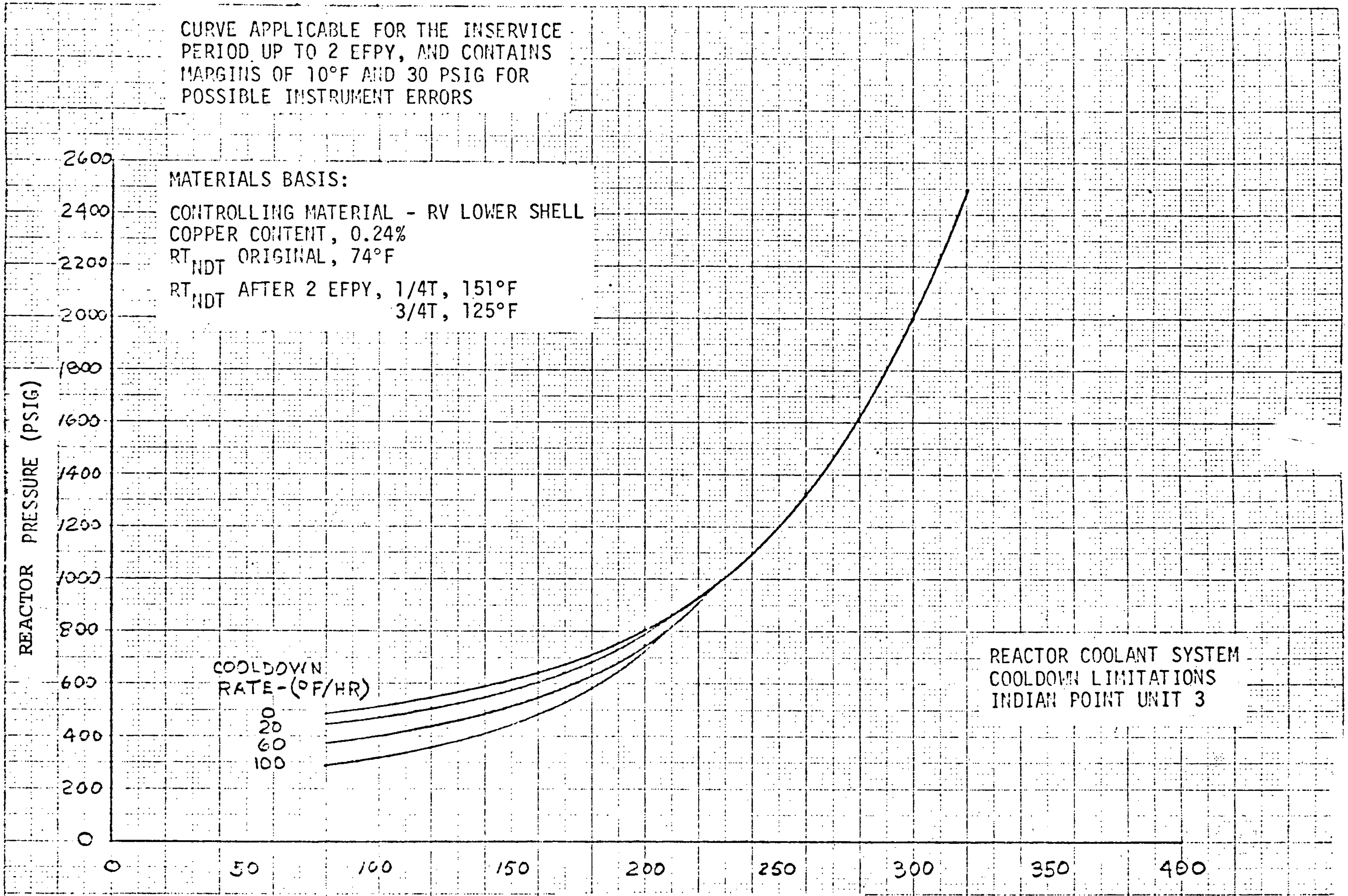
CURVE APPLICABLE FOR THE INSERVICE PERIOD UP TO 2 EFPY, AND CONTAINS MARGINS OF 10°F AND 30 PSIG FOR POSSIBLE INSTRUMENT ERRORS
MATERIALS BASIS:
CONTROLLING MATERIAL - RV LOWER SHELL
COPPER CONTENT, 0.24%
RT NDT ORIGINAL, 74°F
RT NDT AFTER 2 EFPY, 1/4T, 151°F
3/4T, 125°F
REACTOR PRESSURE (PSIG)
2600
2400
2200
2000
1800
1600
1400
1200
1000
800
600
400
200
0
COOLDOWN RATE-(°F/HR)
0
20
60
100
REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS INDIAN POINT UNIT 3
0
50
100
150
200
250
300
350
400
TAVG (°F)

Figure 3.1-2

3.1.E MAXIMUM REACTOR COOLANT OXYGEN, CHLORIDE AND FLUORIDE CONCENTRATION

Specification

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

Containment	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 cannot be controlled within the limits of Specification 3.1.E.1, namely, steady-state limit not restored within 24 hours or transient limit exceeded, 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)	Transients 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, namely, normal concentration not restored within 48 hours or the transient limit exceeded, the reactor shall be immediately brought to the cold shutdown condition and the cause of the out-of-specification condition 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 consistent 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 against stress corrosion cracking 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] during power operation. 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 as the condition can be corrected. Thus the periods of either 24 hours or 48 hours for corrective action to restore concentrations within the limits have been established. If the corrective action has not been effective at the end of the proper period (24 hours or 48 hours), 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 time and 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

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 brought above the cold shutdown condition unless the following requirements are met:

1. Two charging pumps shall be operable. One shall be operating.

2. Two boric acid transfer pumps shall be operable one of which shall be operating to recirculate the contents of the Boron Injection Tank.

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 22,500 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 and a recirculation flow path between

a boric acid tank and the Boron Injection Tank.

5. The boric acid tank level indicators and the Boron Injection Tank recirculation flow indicator shall be operating.

6. Two channels of heat tracing shall be operable for the flow path from the boric acid tanks to the Reactor Coolant System.

7. City water piping and valves shall be operable to the extent required to provide emergency cooling water to the charging pumps and flush water for the concentrated boric acid piping from the outlet of the boric acid storage tanks to the charging pump suction.

C. The requirements of 3.2.B may be modified to allow any one of the following components to be inoperable at any one time:

1. One of the two operable charging pumps may be removed from service provided the standby pump is immediately placed in service and a second charging pump is restored to an operable status within 24 hours.

2. One boric acid transfer pump may be out of service provided the standby pump is immediately placed in service and the failed pump is restored to an operable status within 24 hours.

3. One boric acid storage tank and/or its associated level indicator 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 and/or indicator is restored to an operable status within 48 hours.

4. One channel of heat tracing for the flow path from the boric acid tanks to the Reactor Coolant System may be out of service provided the failed channel is restored to an operable status within 7 days and the redundant channel is demonstrated to be operable daily during that period.

5. The Boron Injection Tank recirculation flow indicator may be inoperable for 48 hours.

D. If the Chemical and Volume Control System is not restored to meet the requirements of 3.2.B within the time period specified in 3.2.C, then:

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

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

3. In either case, if the requirements of 3.2.B are not satisfied within an additional 48 hours, the reactor shall be brought to the cold shutdown condition utilizing normal operating procedures. The shutdown shall start no later than the end of the 48 hour period.

BASIS

The Chemical and Volume Control System[1] 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.

3. Injection of borated water from the boron injection tank and the refueling water storage tank with the safety injection pumps[2]

The Chemical and Volume Control System also provides a means for assuring that the Boron Injection Tank remains filled with borated water having the proper boric acid concentration. This is accomplished by continuously recirculating the contents of the Boron Injection Tank with the contents of one Boric Acid Tank using a Boric Acid Transfer Pump as the driving force and by performing the surveillance requirements detailed in Section 4 of the Technical Specifications.

The quantity of boric acid in storage in 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.

Continuous recirculation between the boric acid storage tanks and the boron injection tank, and operability of the heating tracing circuit on the recirculation line insures that a flow path exists from the boric acid tank to the boron injection tank.

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 channel of heat tracing is sufficient to maintain the specified low temperature limit. The second channel of heat tracing provides backup for continuous plant operation when one channel is inoperable. Should both channels of heat tracing become inoperable, the reactor will be shutdown and can easily be borated before the line temperature is reduced near the boric acid precipitive temperature.

The city water system is used as a source of water for emergency cooling of the charging pumps and as a source of flush water to remove concentrated boric acid from the piping between the outlet of the boric acid storage tanks and the inlet to the charging pumps in the unlikely event of a complete loss of electrical power and/or a complete loss of service water resulting from turbine missiles.

References;

(1) FSAR - Section 6.2

(2) FSAR - Section 9.2

3.6 CONTAINMENT SYSTEM

Applicability

Applies to the integrity of reactor containment.

Objective

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

Specification

a. Containment Integrity

1. The containment integrity (as defined in 1.9) shall not be violated unless the reactor is in the cold shutdown condition. However, those non-automatic valves listed in Table 3.6-1 may be opened if necessary for plant operation and only as long as necessary to perform the intended function.

2. The containment integrity shall not be violated when the reactor vessel head is removed unless the boron concentration is sufficient to maintain the shutdown margin $\geq 10\% \Delta k/k$.

3. If containment integrity requirements are not met when the reactor is above cold shutdown, containment integrity shall be restored within four hours or the reactor shall be brought to a cold shutdown condition within the next 36 hours utilizing normal operating procedures.

b. <u>Internal Pressure</u>

1. If the internal pressure exceeds 2.5 psig or the internal vacuum exceeds 2.0 psig, the condition shall be corrected or the reactor shutdown.

c. <u>Containment Temperature</u>

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

<u>Basis</u>

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 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 6.4 psig.(1) The containment can withstand an internal vacuum of 3 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 linear material.(3)

The following describes the bases for opening the specified non-**automatic** containment isolation valves during periods when containment integrity is required.

1) Valves Required to be Open During a Postulated Accident

   These valves will be open during normal power operation as well as during and/or after the postulated accident. For example, these valves are located in piping of protection, safeguards or essential service systems which will be required to function during and/or after the postulated accident.

2) Valves Normally or Intermittently Open During Normal Plant Operation

   These valves will be open during normal plant operation and are located within piping systems which either service equipment within the containment or are only intermittently opened for periodic testing of safeguards systems, or for calibration purposes. All valves in this category will be manually closed if the postulated accident should occur.

3) Valves Normally Closed for Normal Operation but can be Administratively Opened During Normal Operation

   These valves will be normally closed during normal power operations and for postulated accident conditions.

However, should the need arise for any of these valves to be opened during normal plant operations, they will be administratively opened and administratively closed as required. The probability of a valve in this category being opened and a simultaneous LOCA occurring is extremely small due to the infrequent amount of time the valve is opened.

Even though the probability of a simultaneous LOCA and a valve in this category being opened is extremely small, should this event ever occur, then the valve would be manually closed during the post-LOCA period.

4) Valves Normally Closed for Normal Operation but can be Administratively Opened During Normal Operation and are Required to be Opened Intermittently Following a Postulated Accident

These valves will be normally closed during normal power operations and for postulated accident conditions. However, should the need arise for any of these valves to be opened during normal plant operations, they will be administratively opened and administratively closed as required. In addition, these valves are required to be opened intermittently following a postulated accident.

References

(1) FSAR - Volume 7, Response to Question 14.6

(2) FSAR - Appendix 5A, Section 3.1.8

(3) FSAR - Section 5.1.1.1

TABLE 3.6-1

Non-Automatic Containment Isolation Valves

1) Valves Required to be Open During a Postulated Accident

744	SWN-41 (5 Valves)
869A	SWN-44 (5 Valves)
869B	SWN-51 (5 Valves)
850A	SWN-71 (5 Valves)
851A	PCV-1111 (2 Valves)
752F	1814A
752J	1814B
753F	1814C
753J	

2) Valves Normally or Intermittently Open During Normal Plant Operation

550	250B	859A	863
1870	241B	859C	580A
743	250C	1833A	580B
205	241C	1833B	732
226	250D	1610	958
227	241D	891A	959
250A	878A	891B	990C
241A	878B	891C	885A
		891D	885B

3) Valves Normally Closed for Normal Operation but which can be Administratively Opened During Normal Operation

UH-37
UH-38
SA-24 (2 Valves)

4) Valves Normally Closed for Normal Operation but can be Administratively Opened During Normal Operation and are Required to be Opened Intermittently Following a Postulated Accident

1882A	1890A
1875A	1890B
1875B	1890C
1876A	1890D
1876B	1890E
PS-7	1890F
PS-8	1890G
PS-9	1890H
PS-10	1890J
888A	990A
888B	990B

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, and other operations as specified below , the following conditions shall be satisified:

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. Subsequent to refueling operations, the radiation level inside containment shall be continuously monitored.

3. The core subcritical neutron flux shall be continuously monitored by two operable nuclear source channels, each with continuous visual indication in the control room and one with audible indication in the containment available whenever core geometry is being changed and whenever power is connected to control rod drive mechanisms. The number of operable channels may be reduced to one during performance of tests specified in Specification A.5.A, Table A.5-1,

Item 1. When core geometry is not being changed and power is disconnected from the control rod drive mechanisms, 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 operating except during those core alterations in which the Residual Heat Removal flow interferes with component positioning.

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 containment vent and purge system, including the radiation monitors which initate isolation, shall be tested and verified to be operable within 100 hours prior to refueling operations.

8. Hoists or cranes utilized in handling spent fuel shall be dead-load tested before fuel movement begins. The load assumed by the hoists or cranes for this test must be equal to or greater than the maximum load to be assumed by the hoists or cranes during the refueling operation. A thorough visual inspection of the hoists or cranes shall be

made after the dead load test and prior to fuel handling. A test of interlocks shall also be performed.

B. If any of the specified limiting conditions for refueling are 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 removal 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. A shutdown margin of 10% $\Delta k/k$ in the cold condition with all rods inserted will also maintain the core subcritical even if no control rods were inserted into the reactor.[2] Periodic checks of refueling water boron concentration and residual heat removal pump operation 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 one-hundred hour 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 storage building emergency ventilation system to be operable is established in accordance with standard testing requirements to assure that the system will function to reduce the offsite doses to within acceptable limits in the event of a fuel-handling accident. The system is actuated upon receipt of a signal from the area high activity alarm or by a manually-operated switch. The system is tested prior to fuel-handling and is in a standby basis.

When the spent fuel cask is being placed in or removed from its position in the spent fuel pit, mechanical stops incorporated on the bridge rails 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. 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 5-ton hook, but 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 hoists and cranes before handling spent fuel provide assurance that the hoists or cranes are capable of proper operation.

References

(1) FSAR - Section 9.5.2

(2) FSAR - Table 3.2.1-1

(3) Response to Question 9.1.2, FSAR Volume 7

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 and in accordance with 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.

c) The Reactor Coolant System leak test temperature pressure shall be in accordance with the limits of Figure 4.3-1 for heatup for the first two-year period of operation (2EFPY). Figure 4.3-1 will be recalculated periodically. Allowable pressures during cooldown from the leak test temperature shall be in accordance with Figure 3.1-2.

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 leaktight 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.

The inservice leak test temperatures are shown on Figure 4.3-1. The temperatures are calculated in accordance with ASME Code Section III, Appendix G. This Code requires that a safety factor of 1.5 times the stress intensity factor caused by pressure be applied to the calculation.

For the first 2 effective full-power years, it is predicted that the highest RT_{NDT} in the core region taken at the 1/4 thickness will be 151°F. The temperature determined by methods of ASME Code Section III for 2335 psig is 126°F above this RT_{NDT} and for 2600 psig (maximum) is 136°F above this RT_{NDT}. The minimum inservice leak test temperature requirements for periods up to 2 effective full-power years are shown on Figure 4.3-1.

The heatup limits specified on the heatup curve, Figure 4.3-1 must not be exceeded while the Reactor Coolant System is being heated to the inservice leak test temperature. For cooldown from the leak test temperature, the limitations of Figure 3.1-2 must not be exceeded.

<u>Reference</u>

(1) FSAR Section 4.

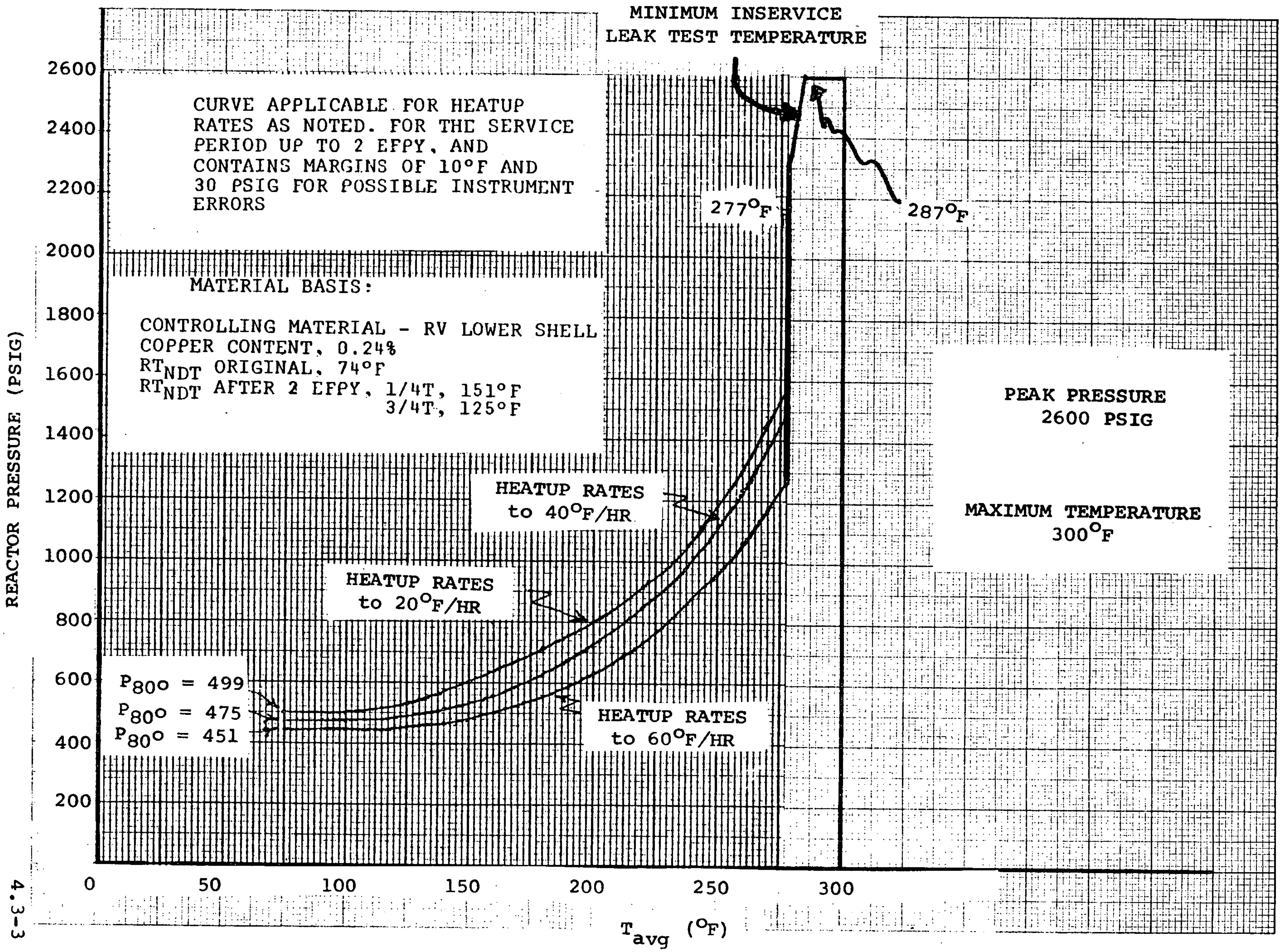

FIGURE 4.3-1 PRESSURE/TEMPERATURE LIMITATIONS FOR HYDROSTATIC LEAK TEST

4.4 CONTAINMENT TESTS

Applicability

Applies to containment leakage.

Objective

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

Specification

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

1. Test

The upper limit for 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 41 psig.

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

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

2. Corrective Action

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

b. If repairs are not completed and conformance to the acceptance criterion is not demonstrated within 7 days, the reactor shall be placed in the cold shutdown condition and all control rods inserted until repairs are effected and the continuous leakage meets the acceptance criterion.

B. Air Lock Tests

1. Whenever containment integrity is required, verification shall be made of proper repressurization to at least 40.6 psig of the double-gasket air lock door seal upon closing an air lock door.

C. Containment Modifications

Any major modification or replacement of components of the containment performed after the initial pre-operational leakage rate test shall be followed by either an integrated leakage rate test, or a local leak detection test and shall meet the appropriate acceptance criteria of 10CFR50 Appendix J. Modifications or replacement performed directly prior to the conduct of an integrated leakage rate test shall not require a separate test.

Basis

The containment is designed for a pressure of 47 psig.(1) While the reactor is operating, the internal environment of the containment will be air at essentially atmospheric pressure and an average maximum temperature of approximately 120°F. With these initial conditions, the temperature of the steam-air mixture at the peak accident pressure of 40.6 psig is 263°F.(4)

Prior to initial operation, the containment was strength-tested at 54 psig and was leak-tested. The acceptance criterion for this pre-operational leakage rate test has been established as 0.075 W/o (.75 L_a) per 24 hours at 40.6 psig and 263°F, which are the peak accident pressure and temperature conditions. This leakage rate is consistent with the construction of the containment,[2] which is equipped with a Weld Channel and Penetration 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 W/o per day for 24 hours. With this leakage rate and with minimum containment engineered safeguards operating, the public exposure would be well below 10CFR100 values in the event of the design basis accident.[3]

The Weld Channel and Penetration System is in service continuously to monitor leakage from potential leak paths such as the containment personnel lock seals and weld channels, containment penetrations containment liner weld channels, double-gasketed seals and spaces between certain containment isolation valves and personnel door locks. During normal plant operation containment personnel lock door seals are continuously pressurized after each closure by the Weld Channel and Penetration Pressurization System.

REFERENCES

(1) FSAR - Section 5

(2) FSAR - Section 5.1.7

(3) FSAR - Section 14.3.5

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 and run for a period of time sufficient to reach stable operating temperatures.
2. At each refueling outage each diesel generator shall be manually started, synchronized and loaded up to its nameplate rating and run for a period of time sufficient to reach operating temperatures.
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 inspected and maintained 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 operates as designed.

B. 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. At each refueling outage each 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 and a 2000 HR rating of 1950 kw. Two diesels can power the minimum safeguards loads.

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 refueling outage load test for each 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.

<u>Reference</u>

FSAR, Section 8.2

A-1 REACTOR COOLANT SYSTEM BORON CONCENTRATION

Applicability

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

Objective

To maintain the core subcritical at all times.

Specification

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

Basis

With a concentration of 2000 ppm boron in the core, there will be a 5% hot shutdown margin and a 6% cold shutdown margin at all times, even if all control rods are fully withdrawn. Maintaining at least the same concentration in the accumulators and in the refueling water storage tank assures that no inadvertent action of the Safety Injection System or the Chemical and Volume Control System could cause dilution of the boron in the Reactor Coolant System.

A-2 CONTROL ROD DRIVES

Applicability

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

Objective

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

Specification

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

Basis

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

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

Applicability

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

Objective

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

Specification

A.3.1 The following valves shall be locked closed during core loading operations:

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

A.3.2 For the purpose of operational testing to comply with Technical Specification requirements,

Valves: 376, 378 may be opened to batch boron.

326, 339 may be opened for blended makeup

363, 365 may be opened under emergency conditions to flush boric acid piping

1132 may be opened to transfer concentrated boric acid from the concentrates holding tank to the boric acid storage tank.

1800, 1841, 873A may be opened to fill the spray additive tank

350 may be opened to fill the refueling water storage tank

1733 may be opened to pump water from Reactor Coolant drain tank to the refueling water storage tank.

Bases

The valves specified in A.3.1 isolate all lines which could conceivably serve as paths for unborated water to the Reactor Coolant System during core loading operations.

The valves listed in A.3.2 are required to be open at certain times following core loading in order to perform required sub-critical testing, and to assure that capability is available to satisfy minimum safeguards requirements such as providing paths for boric acid addition to the reactor coolant system.

A-4 AUXILIARY ELECTRICAL SYSTEMS

Applicability

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

Objective

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

Specification

A. During fuel movement or control rod testing:

1. 2 diesels operable with on-site supply of at least 11,352 gallons of fuel available on-site.

2. 2 batteries plus 2 chargers and the d.c. distribution systems operable corresponding to the 2 operable diesels above.

B. If any of the above conditions are not met:

1. All control rods shall be inserted.

2. All movement of fuel shall be stopped.

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

Basis

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

A battery is required to provide power to required plant instrumentation and start an emergency diesel generator. Since two diesels are required to provide redundancy for electric power availability, redundant batteries are required The actions required in the event of unavailability of a diesel or a battery provide greater assurance that there can be no inadvertant approach to criticality until the diesel or battery is returned to service.

SURVEILLANCE REQUIREMENTS

Operational Safety Review

Applicability

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

Objective

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

Specification

A. Calibration and checking of instrumentation shall be performed as specified in Table A.5-1 except when the parameter to be measured does not exist.

B. Sampling and equipment tests shall be conducted as specified in Tables A.5-2 and A.5-3, respectively except when the parameter to be measured does not exist.

C. The following parameters shall be logged once a shift except when the parameter to be measured does not exist:

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

Basis

Check

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

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

Calibration

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

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

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

TABLE A.5.1

Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels

	Channel Description	Check	Calibrate	Remarks
1.	Nuclear Source Range	S W(1)	N.A.	(1) Instrument Channel Functional Test (Verification of channel response to simulated inputs; prior to and whenever control rods are energized, verify tripping of reactor trip breakers)
2.	Reactor Coolant Temperature	S	R.	
3.	Pressurizer Water Level	S	R	
4.	Pressurizer Pressure (High and Low)	S	R.	
5.	Analog Rod Position	S	R.	
6.	Rod Position Bank Counters	S	N.A.	With analog rod position.
7.	Steam Generator Level	S	R.	
8.	Boric Acid Tank Level	W	R.	Bubbler tube rodded during calibration
9.	Refueling Water Storage Tank Level	W	R.	

TABLE A.5.1
(Continued)

	Channel Description	Check	Calibrate	Remarks
10.	Containment & Fuel Storage Building Area Radiation Monitoring Systems	D	R	

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

S - Each Shift
D - Daily
W - Weekly
N.A. - Not Applicable
R - Each Refueling Outage

TABLE A.5.2

Frequencies for Sampling Tests

		Check	Frequency	Maximum Time Between Tests
1.	Reactor Coolant Samples	Cl & O_2 F	3 Times per 7 Days Weekly	3 Days 10 Days
2.	Reactor Coolant Boron	Boron Concentration	2 Days/wk.	5 Days
3.	Refueling Water Storage Tank Water Sample	Boron Concentration	Monthly	45 Days
4.	Boric Acid Tank	Boron Concentration	Weekly	10 Days
5.	Accumulator	Boron Concentration	Monthly	45 Days

TABLE A.5.3

Frequencies for Equipment Tests

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

A-6 LIMITING SAFETY SETTING, PROTECTIVE INSTRUMENTATION

Applicability

Applies to the trip setting for the source range monitor.

Objective

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

Specification

Whenever power is connected to the control rod drive mechanisms, the source range flux trip setting shall be less than or equal to 2×10^5 counts/sec.

Basis

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

5 DESIGN FEATURES

5.1 SITE

Applicability

Applies to the location and extent of the reactor site.

Objective

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

Specification

The minimum distance from the reactor center line to the boundary of the site exclusion area and the outer boundary of the low population zone as defined in 10 CFR 100.3 is 350 meters[1] and 1100 meters,[2] respectively.

References

(1) FSAR - Section 2.4.4
(2) FSAR - Section 2.4.3

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 0.15g applied horizontally and 0.10g applied vertically at the same time.(1) The containment is also structurally designed to withstand an external pressure 3 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 valves are actuated to the closed position by an automatically derived safety injection signal. A manually initiated containment isolation signal can be generated from the control room to perform the same function. The automatic Phase B containment isolation valves are tripped closed upon actuation of the containment spray system. 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 two internal spray sub-systems each of which is capable of providing a distributed borated water spray of at least 2500 gpm. During the initial period of spray operation, sodium hydroxide would be added to the spray water to increase the removal of iodine from the containment atmosphere.[3]

2. The containment vessel has an internal air recirculation system which includes five fan-cooler units (centrifugal fans and water cooled heat exchangers), each capable of transferring heat at a rate of 21,200 BTU/sec from the containment atmosphere at the post accident design conditions, i.e., a saturated air-steam mixture at 47 psig and 271°F. All of the fan cooler units are equipped with activated charcoal filters to remove volatile iodine following an accident.[4]

References

(1) FSAR Appendix 5A
(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, and reactor coolant system.

Objective

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

A. Reactor Core

1. The reactor core contains approximately 88 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 1434 poison rods in the form of 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 borosilicate 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 11,522 cubic feet.

References

(1) FSAR Section 3.2.2

(2) FSAR Section 3.2.1

(3) FSAR Section 3.2.1

(4) FSAR Section 3.2.3

(5) FSAR Sections 3.2.1 & 3.2.3

(6) FSAR Table 4.1- 10

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.

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Plant Manager shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.

6.2 ORGANIZATION

Facility Management and Technical Support

6.2.1 The organization for Facility Management and Technical Support shall be as shown on Figure 6.2-1.

Facility Staff

6.2.2 The Facility organization shall be as shown on Figure 6.2-2 and:

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

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

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

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

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

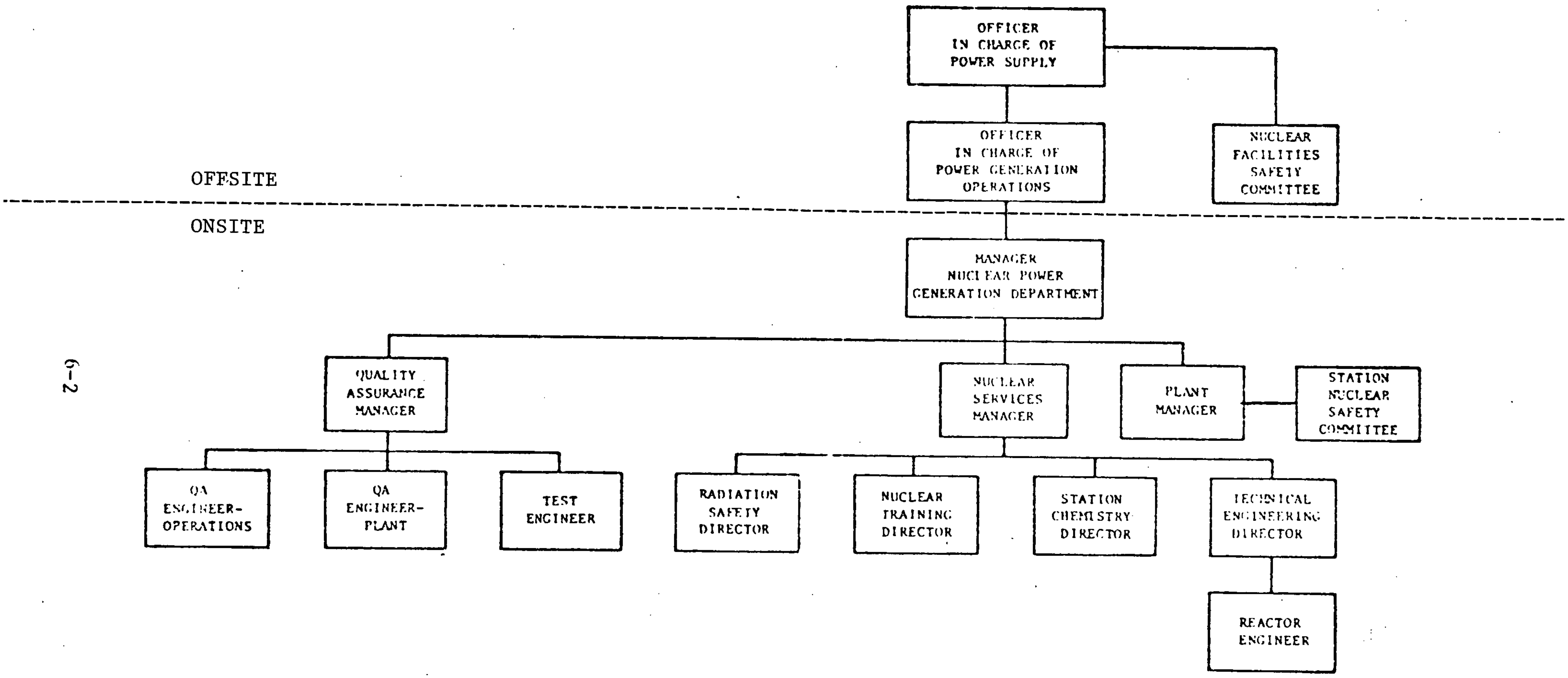

Figure 6.2-1 Facility Management and Technical Support Organization

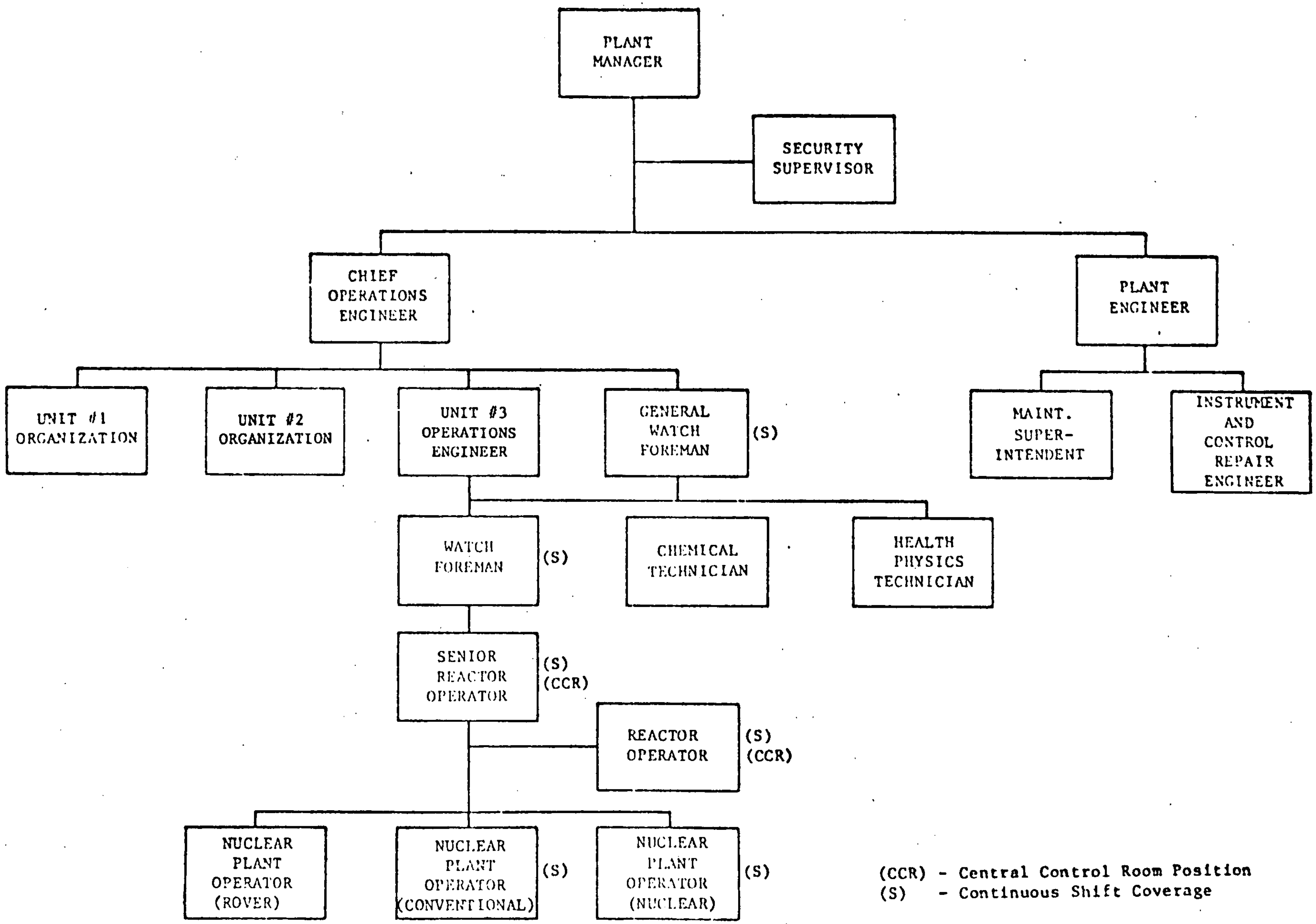

Figure 6.2-2 Facility Organization

Table 6.2-1

Minimum Shift Crew Composition

License Category	During Operations Involving Core Alterations	During Cold Shutdown or Refueling Periods	At All Other Times
Senior Operator License	2*	1	1
Operator License	1	1	2
Non-Licensed	(As Required)	1	2

*Includes individual with SRO license supervising fuel movement as per Section 6.2.2 e.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions.

6.4 TRAINING

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

6.5 REVIEW AND AUDIT

6.5.1 Station Nuclear Safety Committee (SNSC)

Function

6.5.1.1 The Station Nuclear Safety Committee shall function to advise the Plant Manager on all matters related to nuclear safety.

Composition

6.5.1.2 The Station Nuclear Safety Committee shall be composed of the:

Chairman:	Technical Engineering Director
Member:	Radiation Safety Director
Member:	Operations Engineers
Member:	Reactor Engineers
Member:	Station Chemistry Director
Member:	I & C Repair Engineer
Member:	Maintenance Superintendent

Alternates

6.5.1.3 Alternate members shall be appointed in writing by the SNSC Chairman to serve on a temporary basis; however, no more than two alternates shall participate in SNSC activities at any one time.

Meeting Frequency

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

Quorum

6.5.1.5 A quorum of the SNSC shall consist of the Chairman or Vice Chairman and four members including alternates.

Responsibilities

6.5.1.6 The Station Nuclear Safety Committee shall be responsible for:

a. Review of 1) all procedures required by Specification 6.8 and changes thereto, and 2) any other proposed procedures or changes thereto as determined by the Plant Manager to affect nuclear safety.

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

c. Review of all proposed changes to the Technical Specifications.

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

e. Investigation of all violations of the Technical Specifications and preparation and forwarding of a report covering evaluation and recommendations to prevent recurrence via the Plant Manager to the Manager, Nuclear Power Generation Department and to the Chairman of the Nuclear Facilities Safety Committee.

f. Review of facility operations to detect potential safety hazards.

g. Performance of special reviews and investigations and the issuance of reports thereon as requested by the Plant Manager or the Chairman of the Nuclear Facilities Safety Committee.

h. Review of the Plant Security Plan and implementing procedures and submission of recommended changes via the Plant Manager to the Chairman of the Nuclear Facilities Safety Committee.

i. Review of the Emergency Plan and implementing procedures and submission of recommended changes via the Plant Manager to the Chairman of the Nuclear Facilities Safety Committee.

Authority

6.5.1.7 The Station Nuclear Safety Committee shall:

a. Recommend to the Plant Manager, in writing, approval or disapproval of items considered under 6.5.1.6(a) through (d), above.

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

c. Provide immediate written notification to the Chairman, Nuclear Facilities Safety Committee and the Manager, Nuclear Power Generation Department of disagreement between the recommendations of the SNSC and the actions contemplated by the Plant Manager. However, the course of action determined by the Plant Manager pursuant to 6.1.1 above shall be followed.

Records

6.5.1.8 The Station Nuclear Safety Committee shall maintain written minutes of each meeting and copies shall be provided to, as a minimum, the Plant Manager, the Manager, Nuclear Power Generation Department and the Chairman, Nuclear Facilities Safety Committee.

6.5.2 Nuclear Facilities Safety Committee (NFSC)

Function

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

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

Composition

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

The Chairman of the Nuclear Facilities Safety Committee, hereafter referred to as the Chairman, shall be appointed by the Chairman of the Board or the President of the Company.

The Vice Chairman shall be appointed by the Chairman of the Board or the President of the Company. In the absence of the Chairman, he will serve as Chairman.

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

Committee members from departments having a direct interest in nuclear plant design, construction and operation or in nuclear safety shall be designated in writing by the Vice President of the Company who is responsible for the functioning of the department subject to the approval of the Chairman. Committee members from other departments may be appointed by the Chairman with the concurrence of the Vice President of that department.

Alternates

6.5.2.3 Each permanent voting member may appoint an alternate to serve in his absence. Committee records shall be maintained showing each such current designation.

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

Alternate members shall having voting rights.

Consultants

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

Meeting Frequency

6.5.2.5 The NFSC shall meet at least once per calendar quarter during the initial year of facility operation following fuel loading and at least once per six months thereafter.

Quorum

6.5.2.6 A quorum of NFSC shall consist of the Chairman or his designated alternate and a majority of the NFSC members including alternates. In the event both the Chairman and the Vice Chairman are absent, one of the permanent voting members will serve as Acting Chairman. No more than a minority of the quorum shall have line responsibility for operation of the facility.

Review

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

a. The safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

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

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

d. Proposed changes in Technical Specifications or licenses.

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

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

g. ABNORMAL OCCURRENCES, as defined in Section 1.0 of these Technical Specifications.

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

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

<u>Audits</u>

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

a. The conformance of facility operation to all provisions contained within the Technical Specifications and applicable license conditions at least once per year.

b. The performance, training and qualifications of the entire facility staff at least once per year.

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

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

e. The Facility Emergency Plan and implementing procedures at least once per two years.

f. The Facility Security Plan and implementing procedures at least once per two years.

g. Any other area of facility operation considered appropriate by the NFSC or the Senior Company Officer in charge of Power Supply.

Authority

6.5.2.9 The NFSC shall report to and advise the Senior Company Officer in charge of Power Supply on those areas of responsibility specified in Sections 6.5.2.7 and 6.5.2.8.

Records

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

a. Minutes of each NFSC meeting shall be prepared, approved and forwarded to the Senior Company Officer in charge of Power Supply within 14 days following each meeting.

b. Reports of reviews encompassed by Section 6.5.2.7 e, f, g and h, above, shall be prepared, approved and forwarded to the Senior Company Officer in charge of Power Supply within 14 days following completion of the review.

c. Audit reports encompassed by Section 6.5.2.8, above, shall be forwarded to the Senior Company Officer in charge of Power Supply and to the management positions responsible for the areas audited within 30 days after completion of the audit.

6.6 ABNORMAL OCCURRENCE ACTION

6.6.1 The following actions shall be taken in the event of an ABNORMAL OCCURRENCE:

a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Specification 6.9.

b. After each Abnormal Occurrence Report is submitted to the Commission it shall be reviewed by the SNSC and submitted to the NFSC Chairman, the Plant Manager and the Manager, Nuclear Power Generation Department.

6.7 SAFETY LIMIT VIOLATION

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

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

b. The Safety Limit violation shall be reported to the Commission, the Manager, Nuclear Power Generation Department and to the NFSC Chairman immediately.

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

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

6.8 PROCEDURES

6.8.1 Written procedures and administrative policies shall be established, implemented and maintained that meet or exceed the requirements and recommendations of Sections 5.1 and 5.3 of ANSI N18.7-1972 and Appendix A of Regulatory Guide 1.33 except as provided in 6.8.2 and 6.8.3, below.

6.8.2 Each procedure and administrative policy of 6.8.1 above, and any changes to them shall be reviewed and approved for implementation in accordance with a written administrative control procedure approved by the Manager, Nuclear Power Generation Department, with the concurrence of the Station Nuclear Safety Committee and the Nuclear Facilities Safety Committee. The administrative control procedure required by this specification shall, as a minimum, require that:

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

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

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

6.8.3 A mechanism shall exist for making temporary changes, and they shall only be made by approved management personnel in accordance with the requirements of ANSI 18.7-1972. The change shall be documented, and reviewed by the SNSC within 7 days of implementation.

6.9 REPORTING REQUIREMENTS

Routine and Abnormal Occurrence Reports

6.9.1 Information to be reported to the Commission, in addition to the reports required by Title 10, Code of Federal Regulations, shall be in accordance with the Regulatory Position in Revision 3 of Regulatory Guide 1.16, "Reporting of Operating Information - Appendix A Technical Specifications."

Special Reports

6.9.2 Special reports shall be submitted to the Director of Region 1, Office of Inspection and Enforcement, within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. A special report will be prepared covering performance of the Low Pressure Steam Dump System during tests performed at a power level higher than 85% of the licensed application rating. Test results will be extrapolated to verify performance at the design conditions for the license application rating (3025 MWt). The report will be submitted within 90 days of completion of the test.

6.10 RECORD RETENTION

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

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

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

c. ABNORMAL OCCURRENCE REPORTS.

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

e. Records of reactor tests and experiments.

f. Records of changes made to Operating Procedures.

g. Records of radioactive shipments.

h. Records of sealed source leak tests and results.

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

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

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

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

c. Records of facility radiation and contamination surveys.

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

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

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

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

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities required by the QA Manual.

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

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

6.11 RADIATION PROTECTION PROGRAM

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

6.12 RESPIRATORY PROTECTION PROGRAM

Allowance

6.12.1 Pursuant to 10 CFR 20.102(c)(1) and (3), allowance may be made for the use of respiratory protective equipment in conjunction with activities authorized by the operating license for this facility in determining whether individuals in restricted areas are exposed to concentrations in excess of the limits specified in Appendix B, Table I, Column 1, of 10 CFR 20, subject to the following conditions and limitations:

a. The limits provided in 10 CFR 20 Section 20.103(a) and (b) shall not be exceeded.

b. If the radioactive material is of such form that intake through the skin or other additional route is likely, individual exposures to radioactive material shall be controlled so that the radioactive content of any critical organ from all routes of intake averaged over 7 consecutive days does not exceed that which would result from inhaling such radioactive material for 40 hours at the pertinent concentration values provided in Appendix B, Table I, Column 1, of 10 CFR 20.

c. For radioactive materials designated "Sub" in the "Isotope" column of Appendix B, Table I, Column 1 of 10 CFR 20, the concentration value specified shall be based upon exposure to the material as an external radiation source. Individual exposures to these materials shall be accounted for as part of the limitation on individual dose in 10 CFR 20 Section 20.101. These materials shall be subject to applicable process and other engineering controls.

Protection Program

6.12.2 In all operations in which adequate limitation of the inhalation of radioactive material by the use of process or other engineering controls is impracticable, the licensee may permit an individual in a restricted area to use respiratory protective equipment to limit the inhalation of airborne radioactive material, provided:

a. The limits specified in Specification 6.12.1, above, are not exceeded.

b. Respiratory protective equipment is selected and used so that the peak concentrations of airborne radioactive material inhaled by an individual wearing the equipment do not exceed the pertinent concentration values specified in Appendix B, Table I, Column 1, of 10 CFR 20. For the purposes of this subparagraph, the concentration of radioactive material that is inhaled when respirators are worn may be determined by dividing the ambient airborne concentration by the protection factor specified in Table 6.12-1 for the respirator protective equipment worn. If the intake of radioactivity is later determined by other measurements to have been different than that initially estimated, the later quantity shall be used in evaluating the exposures.

c. The licensee advises each respirator user that he may leave the area at any time for relief from respirator use in case of equipment malfunction, physical or psychological discomfort, or any other condition that might cause reduction in the protection afforded the wearer.

d. The licensee maintains a respiratory protective program adequate to assure that the requirements above are met and incorporates practices for respiratory protection consistent with those recommended by ANSI-Z88.2-1969. Such a program shall include:

    1. Air sampling and other surveys sufficient to identify the hazard, to evaluate individual exposures, and to permit proper selection of respiratory protective equipment.

2. Written procedures to assure proper selection, supervision, and training of personnel using such protective equipment.

3. Written procedures to assure the adequate fitting of respirators, and the testing of respiratory protective equipment for operability immediately prior to use.

4. Written procedures for maintenance to assure full effectiveness of respiratory protective equipment, including issuance, cleaning and decontamination, inspection, repair, and storage.

5. Written operational and administrative procedures for proper use of respiratory protective equipment including provisions for planned limitations on working times as necessitated by operational conditions.

6. Bioassays and/or whole body counts of individuals (and other surveys, as appropriate) to evaluate individual exposures and to assess protection actually provided.

e. The licensee shall use equipment approved by the U. S. Bureau of Mines under its appropriate Approval Schedules as set forth in Table 6.12-1. Equipment not approved under U. S. Bureau of Mines Approval Schedules shall be used only if the licensee has evaluated the equipment and can demonstrate by testing, or on the basis of reliable test information, that the material and performance characteristics of the equipment are at least equal to those afforded by U. S. Bureau of Mines approved equipment of the same type, as specified in Table 6.12-1.

f. Unless otherwise authorized by the Commission, the licensee shall not assign protection factors in excess of those specified in Table 6.12-1 in selecting and using respiratory protective equipment.

TABLE 6.12-1 (Page 1 of 5)

PROTECTION FACTORS FOR RESPIRATORS

Description	MODES[1/]	PROTECTION FACTORS[2/] PARTICULATES AND VAPORS AND GASES EXCEPT TRITIUM OXIDE[3/]	GUIDES TO SELECTION OF EQUIPMENT BUREAU OF MINES APPROVAL SCHEDULES* FOR EQUIPMENT CAPABLE OF PROVIDING AT LEAST EQUIVALENT PROTECTION FACTORS *or schedule superseding for equipment of type listed
I. AIR-PURIFYING RESPIRATORS			
Facepiece, half-mask[4/] [7/]	NP	5	21B 30 CFR 14.4(b) (4)
Facepiece, full[7/]	NP	100	21B 30 CFR 14.4(b) (5); 14F 30 CFR 13
II. ATMOSPHERE-SUPPLYING RESPIRATOR			
1. Airline respirator			
Facepiece, half-mask	CF	100	19B 30 CFR 12.2(c) (2) Type C(i)
Facepiece, full	CF	1,000	19B 30 CFR 12.2(c) (2) Type C(i)
Facepiece, full[7/]	D	100	19B 30 CFR 12.2(c) (2) Type C(ii)
Facepiece, full	PD	1,000	19B 30 CFR 12.2(c) (2) Type C(iii)
Hood	CF	5/	6/
Suit	CF	5/	6/

1/, 2/, 3/, 4/, 5/, 6/, 7/ (These notes are on the following pages)

TABLE 6.12-1 (Page 2 of 5)

PROTECTION FACTORS FOR RESPIRATORS

Description	MODES[1/]	PROTECTION FACTORS[2/] PARTICULATES AND VAPORS AND GASES EXCEPT TRITIUM OXIDE[3/]	GUIDES TO SELECTION OF EQUIPMENT BUREAU OF MINES APPROVAL SCHEDULES* FOR EQUIPMENT CAPABLE OF PROVIDING AT LEAST EQUIVALENT PROTECTION FACTORS *or schedule superseding for equipment of type listed
2. Self-contained breathing apparatus (SCBA)			
Facepiece, full[7/]	D	100	13E 30 CFR 11.4(b) (2) (i)
Facepiece, full	PD	1,000	13E 30 CFR 11.4(b) (2) (ii)
Facepiece, full	R	1,000	13E 30 CFR 11.4(b) (1)
III. COMBINATION RESPIRATOR			
Any combination of air-purifying and atmosphere supplying respirator		Protection factor for type and mode of operation as listed above.	19B CFR 12.2(e) or applicable schedules as listed above

1/, 2/, 3/, 4/, 5/, 6/, 7/ (These notes are on the following pages)

1/ See the following symbols:

CF: continuous flow

D: demand

NP: negative pressure (i.e., negative phase during inhalation)

PD: pressure demand (i.e., always positive pressure)

R: recirculating (closed circuit)

2/ a. For purposes of this specification the protection factor is a measure of the degree of protection afforded by a respirator, defined as the ratio of the concentration of airborne radioactive material outside the respiratory protective equipment to that inside the equipment (usually inside the facepiece) under conditions of use. It is applied to the ambient airborne concentration to estimate the concentration inhaled by the wearer according to the following formula:

$$\text{Concentration Inhaled} = \frac{\text{Ambient Airborne Concentration}}{\text{Protection Factor}}$$

b. The protection factors apply:

(i) only for trained individuals wearing properly fitted respirators used and maintained under supervision in a well-planned respiratory protective program.

(ii) for air-purifying respirators only when high efficiency (above 99.9% removal efficiency by U. S. Bureau of Mines type dioctyl phthalate (DOP) test) particulate filters and/or sorbents appropriate to the hazard are used in atmospheres not deficient in oxygen.

(iii) for atmosphere-supplying respirators only when supplied with adequate respirable air.

TABLE 6.12-1 (Page 4 of 5)

3/ Excluding radioactive contaminants that present an absorption or submersion hazard. For tritium oxide approximately half of the intake occurs by absorption through the skin so that an overall protection factor of not more than approximately 2 is appropriate when atmosphere-supplying respirators are used to protect against tritium oxide. Air-purifying respirators are not recommended for use against tritium oxide.

See also footnote 5/, below, concerning supplied-air suits and hoods.

4/ Under chin type only. Not recommended for use where it might be possible for the ambient airborne concentration to reach instantaneous values greater than 50 times the pertinent values in Appendix B, Table I, Column 1 of 10 CFR Part 20.

5/ Appropriate protection factors must be determined taking account of the design of the suit or hood and its permeability to the contaminant under conditions of use. No protection factor greater than 1,000 shall be used except as authorized by the Commission.

6/ No approval schedules currently available for this equipment. Equipment must be evaluated by testing or on basis of available test information.

7/ Only for shaven faces.

NOTE 1: Protection factors for respirators, as may be approved by the U. S. Bureau of Mines according to approval schedules for respirators to protect against airborne radionuclides, may be used to the extent that they do not exceed the protection factors listed in this Table. The protection factors in this Table may not be appropriate to circumstances where chemical or other respiratory hazards exist in addition to radioactive hazards. The selection and use of respirators for such circumstances should take into account approvals of the U. S. Bureau of Mines in accordance with its applicable schedules.

NOTE 2: Radioactive contaminants for which the concentration values in Appendix B, Table I, Column 1, of 10 CFR Part 20 are based on internal dose due to inhalation may, in addition, present external exposure hazards at higher concentrations. Under such circumstances, limitations on occupancy may have to be governed by external dose limits.

Revocation

6.12.3 The specifications of Section 6.12 shall be revoked in their entirety upon adoption of the proposed change to 10 CFR 20, Section 20.103, which would make such provisions unnecessary.

6.13 HIGH RADIATION AREA

6.13.1 As an acceptable alternate to the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20:

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

b. Each High Radiation Area in which the intensity of radiation is greater than 1000 mrem/hr shall be subject to the provisions of 6.13.1(a), above, and in addition locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Watch Foreman on duty.