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January 29, 1976

Re Indian Point Unit No. 3
Docket No. 50-286

Mr. D. B. Vassallo, Chief
Light Water Reactor Branch No. 5
Division of Project Management
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Vassallo

Forwarded herewith please find two hundred (200) copies of technical specifications to be issued as Appendix A to facility operating license DPR-64 for full power operation of Indian Point Unit No. 3.

Very truly yours

Carl L. Newman

Carl L. Newman
Vice President

Enc.
mrb

Sworn to before me this
29 day of January, 1976.

David Watson
Notary Public

DAVID WATSON
Notary Public State of New York
No. 03-4604876
Qualified in Bronx County
Commission Expires March 30, 1977

APPENDIX A

TO

FACILITY OPERATING LICENSE DPR-64

TECHNICAL SPECIFICATION AND BASES

FOR THE

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

WESTCHESTER COUNTY, NEW YORK

POWER AUTHORITY
OF THE STATE
OF NEW YORK

CONSOLIDATED EDISON
COMPANY OF
NEW YORK, INC.

DOCKET NO. 50-286

Date of Issuance:

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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% Δ 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 Technical Specification 3.10 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 Power Operation Condition

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

1.2.5 Refueling Operation Condition

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

1.3 REFUELING OUTAGE

An outage in which core alterations are performed in order to compensate for fuel burnup.

1.4 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.

1.5 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.

1.6 OPERATING/INSERVICE

Performing the intended functions in the intended manner.

1.7 PROTECTION INSTRUMENTATION AND LOGIC

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

1.7.1 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.

1.7.2 Logic Channel

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

1.8 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.

1.9 INSTRUMENTATION SURVEILLANCE

1.9.1 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.

1.9.2 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.

1.9.3 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.

1.9.4 Logic Channel Functional Test

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

1.10 CONTAINMENT INTEGRITY

Containment integrity is defined to exist when:

1.10.1 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.

1.10.2 The equipment door is properly closed.

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

1.10.4 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.

1.10.5 The containment leakage satisfies Specification 4.4.

1.11 REPORTABLE OCCURRENCE

A reportable occurrence shall be any of those conditions specified in Revision 4 of Regulatory Guide 1.16, "Reporting of Operating Information - Appendix A Technical Specifications."

1.11.1 Events requiring prompt notification of the NRC in accordance with Section C.2a of Revision 4 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 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 sub-critical, 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 manmade 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 4 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 QUADRANT POWER TILT RATIO

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

1.13 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 eight months of the previous refueling outage, the surveillance testing need not be performed. The maximum interval between surveillance tests is eighteen 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 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMIT, REACTOR CORE

Applicability

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

Objective

To maintain the integrity of the fuel cladding.

Specification

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

Basis

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all operating conditions. This is accomplished by operating the hot region of the core within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters:

thermal power, reactor coolant temperature and pressure have been related to DNB through the W-3 DNB "L" grid geometry correlation.[3] The W-3 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.30. This corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.[1]

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

The calculation of these limits includes an $F^N_{\Delta H}$ of 1.55, DNB penalties for increased pellet eccentricity, local power spikes, 8% uncertainty in $F^N_{\Delta H}$, and a reference cosine with a peak of 1.55 for axial power shape.[3]

Figures 2.1-1 and 2.1-2 include an allowance for an increase in the enthalpy rise hot channel factor at reduced power based on the expression:

$F^N_{\Delta H} = 1.55\ [1 + 0.2\ (1-P)]$ where P is the fraction of rated power.[3]

The control rod insertion limits are covered by Specification 3.10. Higher hot channel factors could occur at lower power levels because additional control rods are in the core. However, the control rod insertion limits for four loop and three loop operation as dictated by Figures 3.10-4 and 3.10-5, respectively, insure that the DNBR is always greater at partial power than at full power.[3]

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

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

References

[1] FSAR Section 3.2.2

[2] FSAR Section 3.2.1

[3] WCAP-8147, "Fuel Densification-Indian Point Nuclear Generating Unit No. 3", July, 1973; Westinghouse Non-Proprietary Class 3.

[4] FSAR Section 14.1.1

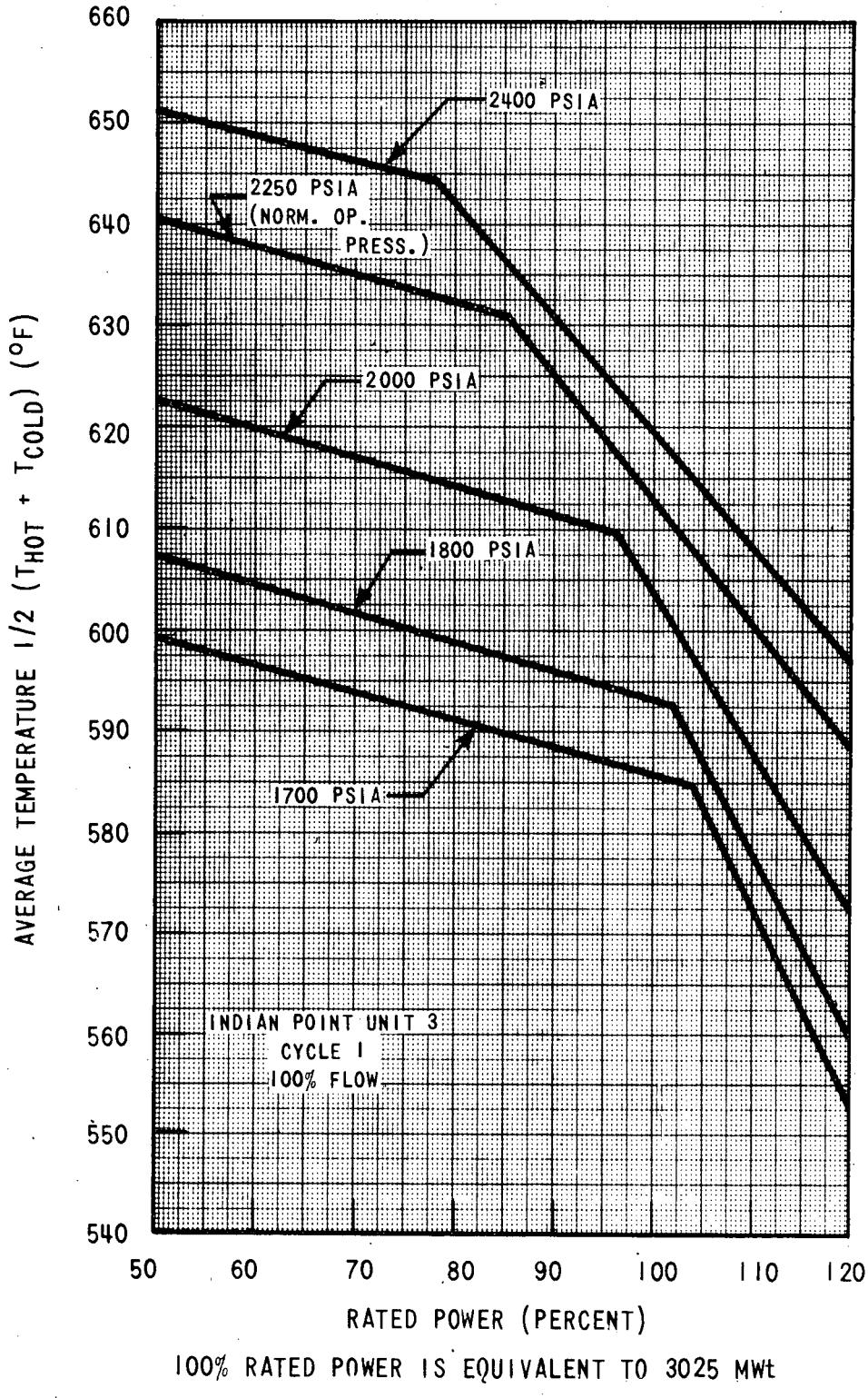


Figure 2.1-1. Core Limits - Four Loop Operation

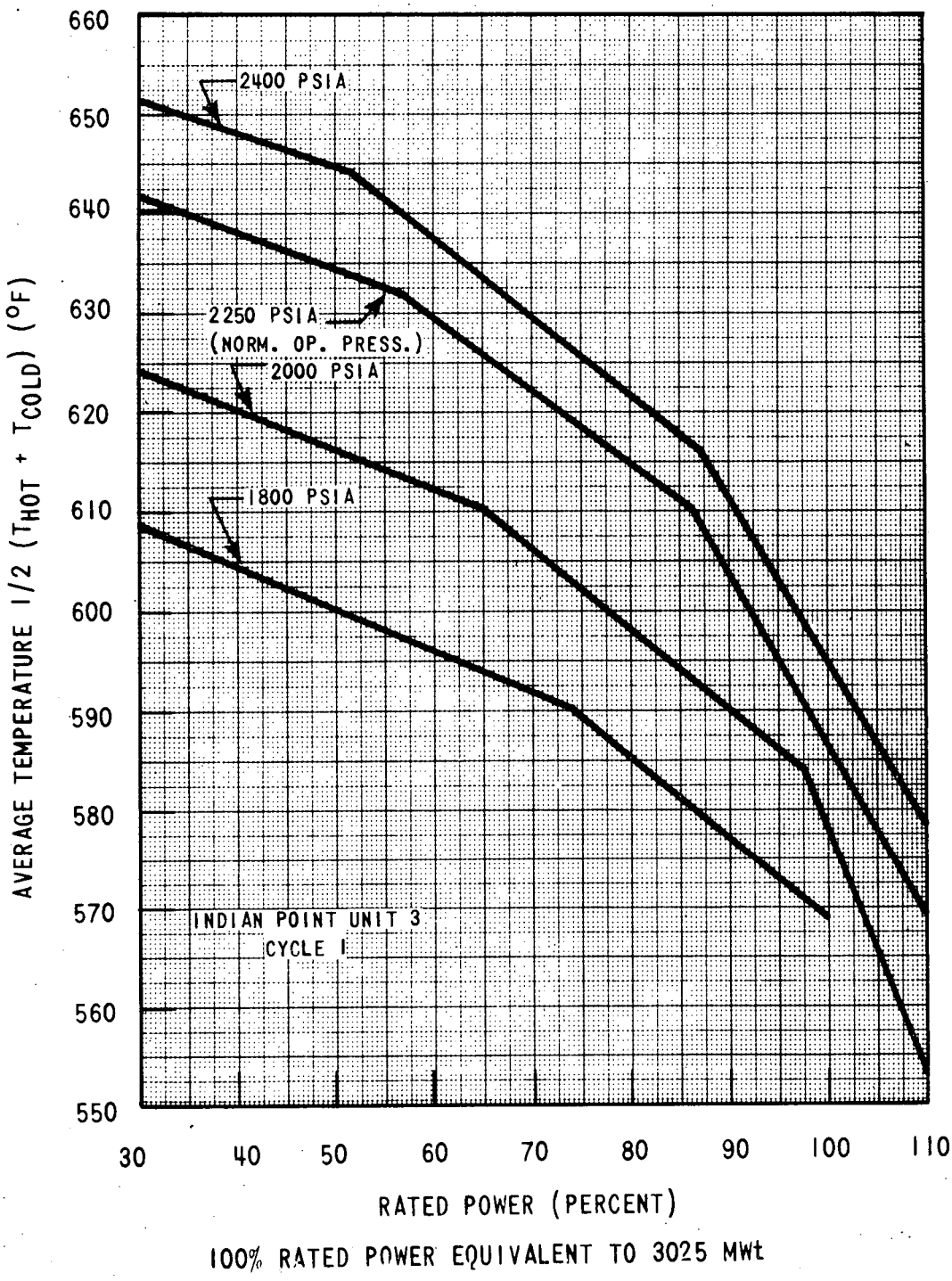


Figure 2.1-2. Core Limits - Three Loop Operation

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.

References

(1) FSAR Section 4

(2) FSAR Table 4.1-1

(3) FSAR Section 4.3.4

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Applicability

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

Objective

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

Specification

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

   A. Startup protection

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

   B. Core limit protection

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

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

      (3) Low pressurizer pressure - $\geq$1800 psig.

      (4) Overtemperature ΔT

$$\Delta T \leq \Delta T_o \left[K_1 - K_2 (T_{avg} - T') + K_3 (P - P') - f(\Delta I)\right]$$

where

ΔT_o = Indicated ΔT at rated power, 57.8°F

T_{avg} = Average temperature, °F

T' = Indicated T_{avg} at nominal conditions at rated power, 571.5°F

P = Pressurizer pressure, psig

P' = Indicated nominal pressurizer pressure at rated power = 2235 psig

$K_1 \leq 1.200$ $K_2 \geq 0.0129$ $K_3 \leq 0.00073$	Four Loop Operation	$K_1 \leq 1.110$ $K_2 \geq 0.0129$ $K_3 \leq 0.00073$	Three Loop Operation

K_1 is a constant which defines the over temperature ΔT trip margin during steady state operation if the temperature, pressure and $f(\Delta I)$ terms are zero.

K_2 is a constant which defines the dependence of the overtemperature ΔT set point to T_{avg}.

K_3 is a constant which defines the dependence of the overtemperature ΔT set point to pressurizer pressure.

ΔI = $q_t - q_b$, where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of rated power.

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

(a) for $q_t - q_b$ within -20, +16 percent, $f(\Delta I) = 0$.

(b) for each percent that the magnitude of $q_t - q_b$ exceeds +16 percent, the ΔT trip set point shall be automatically reduced by an equivalent of 6.0 percent of rated power.

(c) for each percent that the magnitude of $q_t - q_b$ exceeds -20 percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of 1.5 percent of rated power.

(5) Overpower ΔT

$$\Delta T \leq \Delta T_o \left[K_4 - K_5 \frac{dT_{avg}}{dt} - K_6 (T_{avg} - T') - f(\Delta I)\right]$$

where

ΔT_o = indicated ΔT at rated power, °F

T_{avg} = average temperature, °F

T' = indicated T_{avg} at nominal conditions at rated power, 571.5°F

K_4 ≤ 1.09

K_5 = 0 for decreasing average temperature

≥ 0.175 sec/°F for increasing average temperature

K_6 = 0 for $T \leq T'$

≥ 0.00127 for $T > T'$

K_4 is a constant which defines the overpower ΔT trip margin during steady state operation if the temperature and the $f(\Delta I)$ terms are zero.

K_5 is a constant determined by dynamic considerations to compensate for piping delays from the core to the loop temperature detectors; it represents the combination of the equipment static gain setting and the time constant setting.

K_6 is a constant which defines the dependence of the overpower ΔT setpoint to T_{avg}.

$f(\Delta I)$ = as defined above.

$\frac{dT_{avg}}{dt}$ = rate of change of T_{avg}

(6) Low reactor coolant loop flow:

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

(b) Low reactor coolant pump frequency - $\geq$ 55.0 cps

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

C. Other reactor trips

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

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

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

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

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

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

B. The single loop loss of flow reactor trip may be bypassed when the power range nuclear instrumentation indicates $\leq$50% of rated power. The single loop loss of flow reactor trip may be bypassed below 75% of rated power only after the overtemperature ΔT trip setpoint has been adjusted to the three-loop operation value given in 2.3.1.B(4) above. The single loop loss-of-flow trip setpoint is hereafter referred to as P-8.

Basis

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

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

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

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

The overtemperature Delta-T reactor trip provides core protection against DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided only that (1) the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 3.5 seconds)[5], and (2) pressure is within the range between the high and low pressure reactor trips. With normal axial power distribution, the reactor trip limit, with allowance for errors[2], is always below the core safety limit as shown on Figure 2.1-1. If axial peaks are greater than design, as indicated by difference between top and bottom power range nuclear detectors, the reactor trip limit is automatically reduced.[6][7] The values of the constants K_1, K_2, and K_3 are determined during the design of the core, both for operation with all reactor loops in service and for operation with one loop out of service. The values are then specified for the reactor protection system manufacturer and for calibration.

In order to operate with a reactor coolant loop out of service (three-loop operation) no reactor protection system setpoints require modification. The four-loop operation P-8 setpoint is selected to protect against DNB during three-loop operation exclusive of the overtemperature ΔT setpoints. Sustained operation with a reactor coolant loop out of service is expected to be an infrequent occurance. If operation above the four-loop P-8 setpoint is desired, the overtemperature ΔT channels may be

recalibrated using the three loop operation values of K_1, K_2, and K_3, after which the P-8 setpoint may be raised to its three loop operation value. These adjustments and calibrations must be made in the protection system racks and are performed as is done for four-loop operation using calibration procedures. The setpoint adjustments are made based on limits for reduced power three-loop operation and provide sufficient margin for three-loop operations.

The overpower Delta-T reactor trip prevents power density anywhere in the core from exceeding 112% of design power density, as described in Section 7.2.3 and 14.1.2 and includes corrections for axial power distribution, change in density and heat capacity of water with temperature, and dynamic compensation (via the overall gain in the rate controller) for piping delays from the core to the loop temperature detectors. The specified set points meet this requirement and include allowance for instrument errors.[2] The values of the constants K_4, K_5, and K_6 are determined during the design of the core and the reactor protection system. The values are then specified for the reactor protection system manufacturer and for calibration.

The overpower and overtemperature protection system setpoints include the effects of fuel densification on core safety limits. The revised setpoints as given above will ensure that the combination of power, temperature, and pressure will not exceed the revised core safety limits as shown in Figures 2.1-1 and 2.1-2.[10]

The overpower limit criteria is that core power be prevented from reaching a value at which fuel pellet centerline melting would occur. Fuel temperature decreases due to cladding creepdown with burnup and consequential reduction of pellet-cladding gap. Thus overpower limits become less restrictive as fuel burnup proceeds.

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

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

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

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

Above 10% power, an automatic reactor trip will occur if two reactor coolant pumps are lost during operation. Above the P-8 setpoint for four-loop operation, an automatic reactor trip will occur if any pump is lost. This latter trip will prevent the minimum value of the DNB ratio, DNBR, from going below 1.30 during normal operational transients and anticipated transients when only three loops are in operation and the overtemperature ΔT trip setpoint is adjusted to the value specified for four loop operation. When the overtemperature ΔT trip and P-8 setpoints are adjusted to the values specified for three loop operation, the P-8 setpoint for three-loop operation will prevent the minimum value of the DNBR from going below 1.30 during normal operational transients and anticipated transients when only three loops are in operation.

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

References

(1) FSAR 14.1.1

(2) FSAR 14.1.2

(3) FSAR Table 14-1

(4) FSAR 14.3.1

(5) FSAR 14.1.2

(6) FSAR 7.2

(7) FSAR 3.2.1

(8) FSAR 14.1.6

(9) FSAR 14.1.9

(10) WCAP-8147, "Fuel Densification-Indian Point Nuclear Generating Unit No. 3", July 1973; Westinghouse Non-Proprietary Class 3.

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.

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

   c. Reactor power shall not be increased above 50% of rated power with only three pumps in operation unless the overtemperature ΔT trip setpoint for three loop operation has been set in accordance with Specification 2.3.1.B(4).

2. 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 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 ± 1% allowance for error.

Basis

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

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

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

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

The combined capacity of the three pressurizer safety valves is greater than the maximum surge rate resulting from complete loss of load[2] without a direct reactor trip or any other control.

References

1) FSAR Section 14.1.6
2) FSAR Section 14.1.8

B. HEATUP AND COOLDOWN

Specifications

1. The reactor coolant temperature and pressure and system heatup and cooldown rates averaged over one hour (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1-1 and Figure 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. Figure 3.1-1 and Figure 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 Figure 3.1-1 and Figure 3.1-2 shall be recalculated periodically using methods discussed in the Basis and results of surveillance specimens as covered in Specification 4.2.

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 one 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 integrity 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 [6] and ASTM E185[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[1], and the calculation methods described in WCAP-7924.[2]

Heatup and cooldown limit curves are calculated using the most limiting value of RT_{NDT} at the end of two years of service life. The two-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 Figure 4.2-10[4]. Using the applicable fluence at the end of the two-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 G in Section III of the ASME Boiler and Pressure Vessel Code and discussed in detail in WCAP-7924.[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[1] for the metal temperature at that time. Furthermore, the approach applies an 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:

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

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 on 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 at 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 WCAP-7924 [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.

REFERENCES:

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.

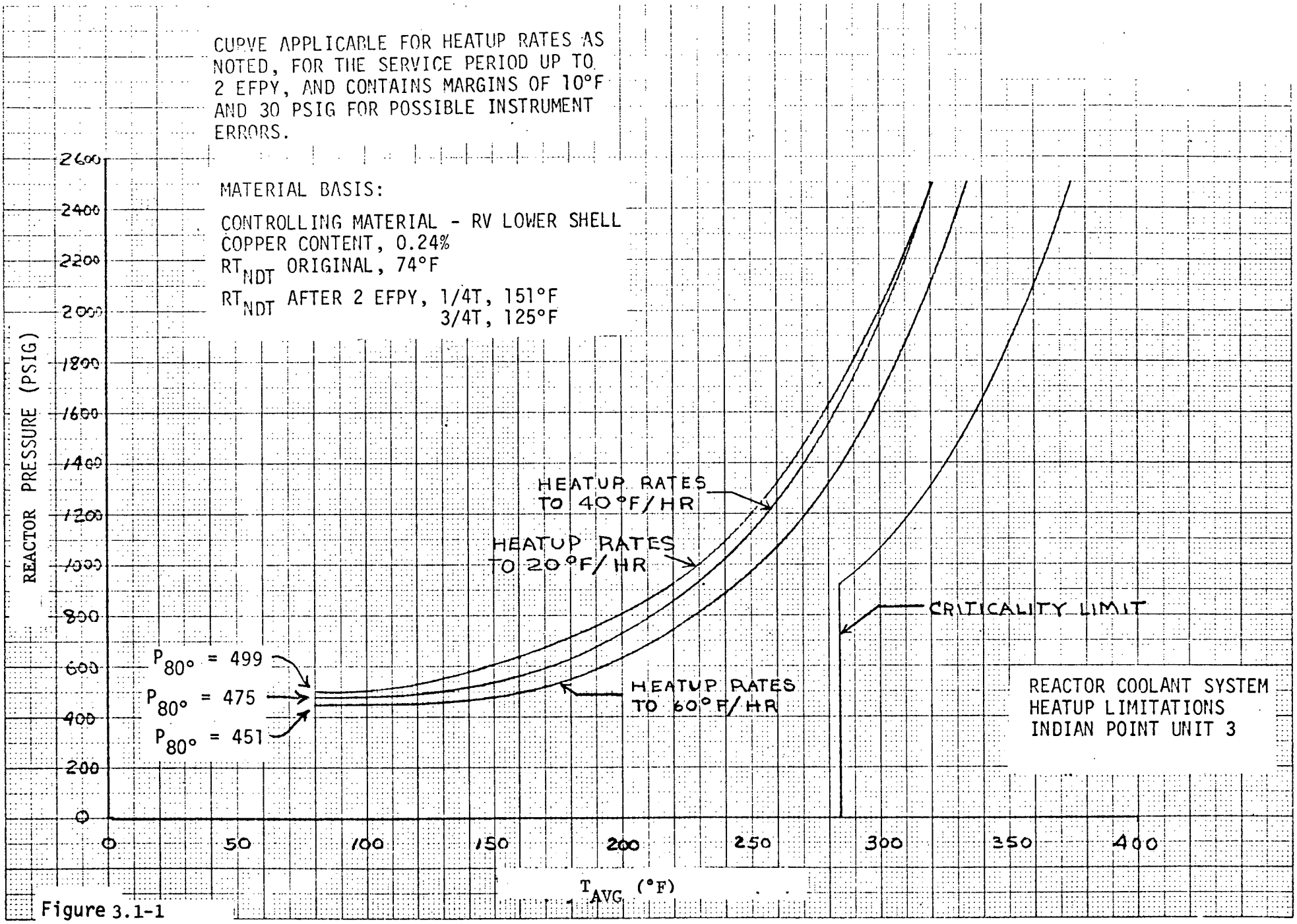

CURVE APPLICABLE FOR HEATUP RATES AS NOTED, FOR THE SERVICE PERIOD UP TO 2 EFPY, AND CONTAINS MARGINS OF 10°F AND 30 PSIG FOR POSSIBLE INSTRUMENT ERRORS.
MATERIAL BASIS:
CONTROLLING MATERIAL - RV LOWER SHELL
COPPER CONTENT, 0.24%
RT NDT ORIGINAL, 74°F
RT NDT AFTER 2 EFPY, 1/4T, 151°F
3/4T, 125°F
REACTOR PRESSURE (PSIG)
2600
2400
2200
2000
1800
1600
1400
1200
1000
800
600
400
200
0
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
0
50
100
150
200
250
300
350
400
T AVG (°F)

Figure 3.1-1

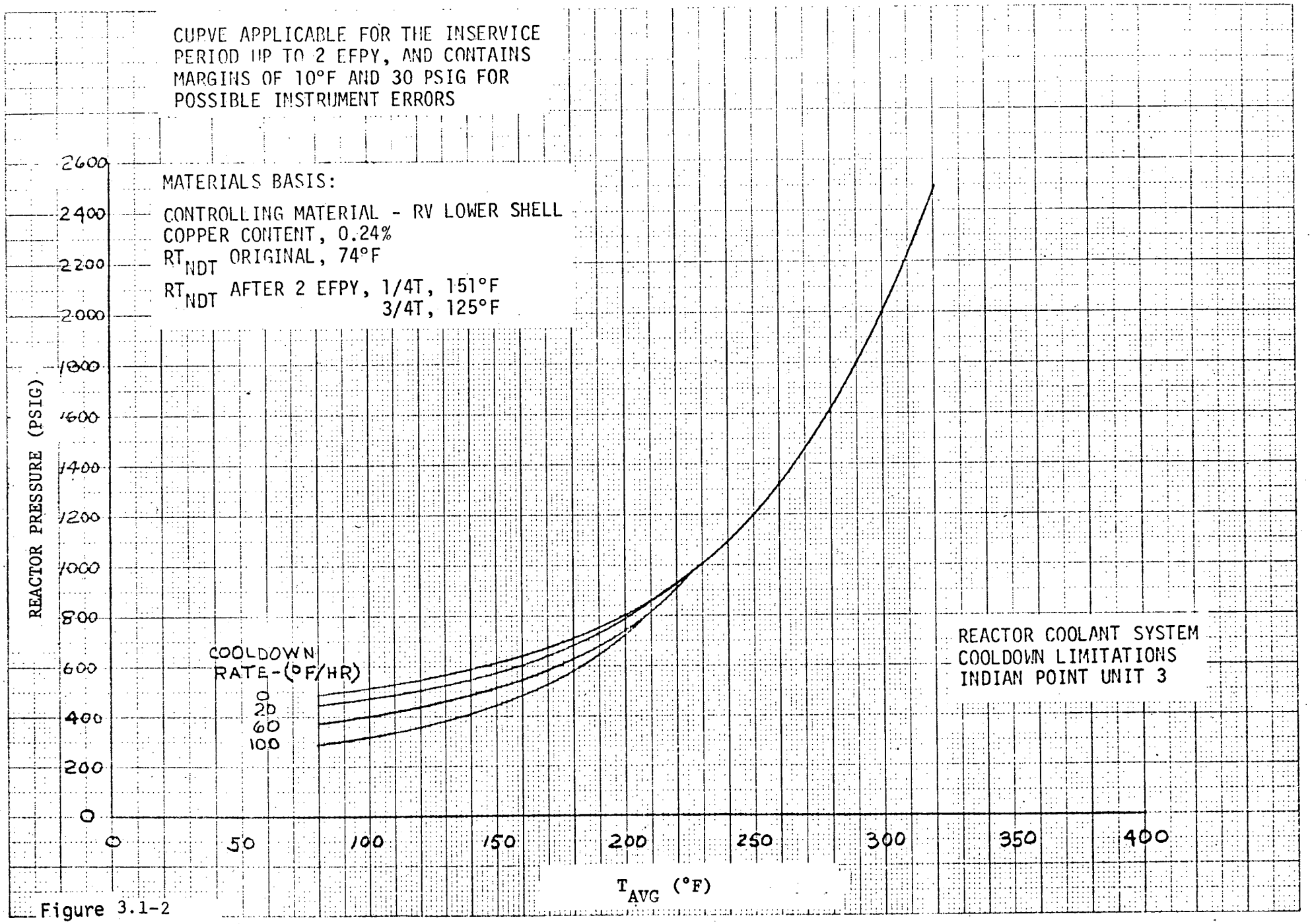

CURVE APPLICABLE FOR THE INSERVICE PERIOD UP TO 2 EFPY, AND CONTAINS MARGINS OF 10°F AND 30 PSIG FOR POSSIBLE INSTRUMENT ERRORS
MATERIALS BASIS:
CONTROLLING MATERIAL - RV LOWER SHELL
COPPER CONTENT, 0.24%
RTNDT ORIGINAL, 74°F
RTNDT AFTER 2 EFPY, 1/4T, 151°F
3/4T, 125°F
COOLDOWN RATE-(°F/HR)
0
20
60
100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

Figure 3.1-2

C. MINIMUM CONDITIONS FOR CRITICALITY

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

2. The reactor shall not be brought to a critical condition until the pressure temperature state is to the right of the criticality limit line shown in Figure 3.1-1.

3. The reactor shall be maintained subcritical by at least $1\%\frac{\Delta k}{k}$ until normal water level is established in the pressurizer.

Basis:

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

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

The requirement that the reactor is not to be made critical except in accordance with Figure 3.1-1 provides increased assurance that the proper relationship between reactor coolant pressure and temperature will be maintained during system heatup and pressurization whenever the reactor vessel is in the nil-ductility temperature range. Heatup to this temperature will be accomplished by operating the reactor coolant pumps.

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

References:

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

D. Primary Coolant Activity

Specification

1. Whenever the reactor is critical or the average reactor coolant temperature is > 500°F, the specific activity of the primary coolant shall be limited to:

   a. $\leq$ 1.0μCi/cc Dose Equivalent I-131, and
   b. $\leq$ 100/$\overline{E}$μCi/cc for all noble gases with half-lives greater than 10 minutes

2. If the specific activity of the primary coolant is > 1.0 μCi/cc Dose Equivalent I-131 but within the allowable limit (below and to the left of the line) shown on Figure 3.1-3, operation may continue for up to 48 hours. The cumulative time for operation under these circumstances shall not exceed 10 percent of the unit's total yearly operating time.

3. If the specific activity of the primary coolant is > 1.0 μCi/cc Dose Equivalent I-131 for more than 48 hours during one continuous time interval or exceeds the limit line shown on Figure 3.1-3, the reactor shall be immediately brought to the hot shutdown condition with $T_{avg} \leq$ 500°F utilizing normal operating procedures.

4. If the specific activity of the primary coolant is > 100/$\overline{E}$ μCi/cc for all noble gases with half-lives greater than 10 minutes, the reactor shall be immediately brought to the hot shutdown condition with $T_{avg} \leq$ 500°F utilizing normal operating procedures.

5. If the specific activity of the primary coolant is > 1.0 μCi/cc Dose Equivalent I-131 or > 100/$\overline{E}$μCi/cc for all noble gases with half-lives greater than 10 minutes, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2. This report shall contain the results of the specific activity analyses, together with the following information:

a. reactor power history starting 48 hours prior to the first sample in which the limit was exceeded.

b. fuel burnup by core region.

c. cleanup flow history starting 48 hours prior to the first sample in which the limit was exceeded.

d. history of de-gassing operations, if any, and

e. the time duration when the specific activity of the primary coolant exceeded 1.0μCi/cc Dose Equivalent I-131.

Bases

The limitations on the specific activity of the primary coolant insure that the resulting 2-hour doses at the site boundary will not exceed 1.5 rem to the thyroid and 0.5 rem whole body following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM and a resultant loss of offsite power. Accident meteorological conditions (5% X/Q) are assumed to exist.

The action statement permitting Power Operation to continue for limited time periods with the primary coolant's specific activity > 1.0μCi/cc Dose Equivalent I-131, but within the allowable limit shown on Figure 3.1-3, accommodates possible iodine spiking phenomenon which may occur following changes in Thermal Power. Operation with specific activity levels exceeding 1.0μCi/cc Dose Equivalent I-131 but within the limits shown on Figure 3.1-3 must be restricted to no more than 10 percent of the unit's yearly operating time, since the

activity levels allowed by Figure 3.1-3 could significantly increase the two-hour thyroid dose at the site boundary following a postulated steam generator tube rupture.

Reducing T_{avg} to < 500°F prevents the release of activity, should a steam generator tube rupture, since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Increased surveillance for performing isotopic analyses for iodine is required whenever the Dose Equivalent I-131 exceeds 1.0 μCi/cc and following a significant change in power level to monitor possible iodine spiking phenomenon.

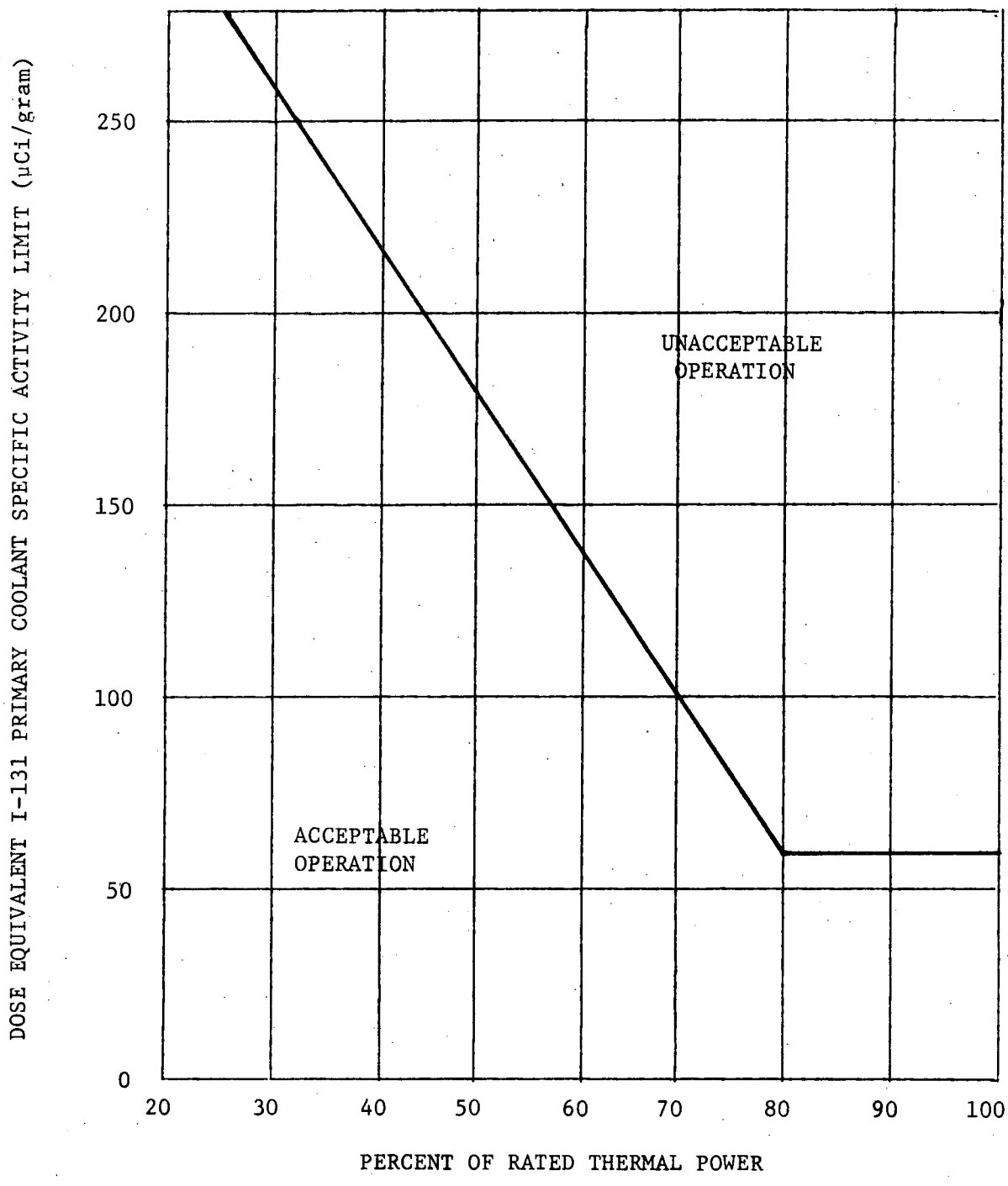


FIGURE 3.1-3

DOSE EQUIVALENT I-131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER with the Primary Coolant Specific Activity > 1.0μCi/gram Dose Equivalent I-131

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)	Transient not to exceed 48 hours (PPM)
a. Oxygen	Saturated	Saturated
b. Chloride	0.15	1.5
c. Fluoride	0.15	1.5

If the limits above are exceeded, namely, normal concentration limits not restored within 48 hours or the transient limits 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

F. LEAKAGE OF REACTOR COOLANT

Specification

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

2. If the leakage rate, excluding controlled leakage sources such as the Reactor Coolant Pump Controlled Leakage Seals and Leakage into Closed Systems, exceeds 1 gpm and the source of leakage is not identified with twenty-four hours of detection, the reactor shall be brought to hot shutdown within four hours. If the source of leakage is not identified within an additional twenty-four hours, the reactor shall be brought to a cold shutdown condition within the next twenty-four hours.

3. If the sources of leakage are identified and the results of the evaluation are that continued operation is safe, operation of the reactor with a total leakage, other than from controlled sources or into closed systems, not exceeding 10 gpm shall be permitted except as specified in 3.1.F.4 below.

4. If it is determined that leakage exists through a non-isolable fault which has developed in a Reactor Coolant System Component Body, pipe wall (excluding steam generator tubes), vessel wall or pipe weld, the reactor shall be brought to the cold shutdown condition within twenty-four hours.

5. If the total leakage, other than from controlled sources or into closed systems, exceeds 10 gpm or if the leakage through any one steam generator exceeds 500 gallons per day or the total leakage through all four steam generators exceeds 1.0 gpm. the reactor shall be placed in the hot shutdown condition within four hours and the cold shutdown condition within an additional twenty-four hours.

6. The reactor shall not be restarted following shutdown as per items 3.1.F.2, 3, 4 or 5, above, until the leak is repaired or until the problem is otherwise corrected.

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

Basis:

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

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

The distinction between identified and unidentified leakage in the specification is made because once the leakage source is identified, the seriousness can be easily evaluated. The strict limit of 1 gallon per minute for unidentified leakage is adopted because in the worst case the leakage source may increase with time or the coolant may impinge on or accumulate in a critical component.

When the source of leakage has been identified, the situation can be evaluated to determine if operation can safely continue. This evaluation will be performed by the Watch Force. Under these conditions, an allowable primary system leakage rate of 10 gpm has been established. This explained leakage rate of 10 gpm is also well within the capacity of one charging pump and makeup would be available even under the loss of off-site power condition.

Controlled sources of reactor coolant system leakage are sources which are designed to leak at a controlled rate. For example, the reactor coolant pump seals are controlled leakage sources. Leakage through a valve packing or a closed valve is not considered as controlled leakage. Leakage into closed systems is that leakage which can be accounted for and contained by a system not directly connected to the atmosphere. Leakage past the pressurizer safety valve seats and steam generator tube leakage are examples of reactor coolant system leakage into closed systems.

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

a. The containment air particulate monitor (R-11).

b. The containment radiogas monitor (R-12).

c. The containment humidity detectors.

d. A leakage detection system which determines leakage losses from all water and steam systems within the containment. This system collects and measures moisture condensed from the containment atmosphere by cooling coils of the main air recirculation units.

The most sensitive and rapid method for detecting small amounts of Reactor Coolant System leakage is the monitoring of the containment airborne radioactivity. Containment gaseous and particulate activity is continuously, automatically monitored. The leakage rate can be determined by the relationship of the airborne activity to the reactor coolant activity.

Measurement of the leakage rate to the containment atmosphere is also possible through humidity detection and condensation collection and measurement. However, it is expected that the containment activity method will give the initial indication of coolant leakage. The other methods will be employed primarily to confirm that leakage exists, to indicate the location of the leakage sources, and to measure the leakage rate.

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

Total reactor coolant leakage can be determined by means of periodic water inventory balances. If leakage is into another closed system, it will be detected by the plant radiation monitors and/or inventory control.

Twenty-four hours is allowed from the time of leakage detection to identify the leakage source and to measure the leakage rate. This time period is required since identification and quantification of leakage sources of less than ten gallons per minute require a careful gathering and evaluation of data and/or a visual inspection of the reactor coolant system.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those limits found to result in negligible corrosion of the steam generator tubes. If stress corrosion cracking occurs, the extent of cracking during plant operation would be limited by the limitation of steam generator leakage between the primary coolant system and the secondary coolant system. Cracks having a primary-to-secondary leakage less than 500 gallons per day during operation will have an

adequate margin of safety against failure due to loads imposed by design basis accidents. Operating plants have demonstrated that primary-to-secondary leakage as low as 0.1 gpm will be detected. Leakage in excess of 500 gallons per day per steam generator or 1 gpm total for all four steam generators will require plant shutdown and an unscheduled eddy current inspection, during which the leaking tubes will be located and plugged. The 500 gallon per day per steam generator limit is also consistent with the assumptions used to develop the Technical Specification limit on secondary coolant activity.

References

FSAR Sections 11.2.3 and 14.2.4

G. Secondary Coolant Activity

Specification

1. Whenever the average reactor coolant temperature is $\geq$ 350°F, the specific activity of the secondary coolant system shall be $\leq$ 0.10 μCi/gram of Dose Equivalent I-131.

2. If the specific activity of the secondary coolant system exceeds 0.10 μCi/gram of Dose Equivalent I-131, the reactor shall be immediately brought to the hot shutdown condition with T_{avg} < 350°F utilizing normal operating procedures.

Basis

The limitations on secondary system specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction of 10CFR Part 100 limits in the event of a steam line rupture. The restriction of 0.1 μCi/gram Dose Equivalent I-131 in the secondary system limits the two-hour thyroid exposure dose to 1.5 rem at the site boundary under these accident conditions. This accident analysis also includes the effects of a coincident 500 gallons per day primary to secondary tube leak in the steam generator of the affected steam line and considers the effect of a coincident iodine spike. Accident meteorological conditions are assumed (5% X/Q) and a decontamination factor of 10 is applied between the water and steam phases.

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.

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 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 heat tracing circuit of 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 9.2
2) FSAR - Section 6.2

3.3 ENGINEERED SAFETY FEATURES

Applicability

Applies to the operating status of the Engineered Safety Features.

Objective

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

Specification

The following specifications apply except during low temperature physics tests.

A. Safety Injection and Residual Heat Removal Systems

1. The reactor coolant system T_{avg} shall not exceed 200°F unless the following requirements are met:

   a. The refueling water storage tank contains a minimum of 346,870 gallons of water at a boron concentration of at least 2000 ppm.

   b. One refueling water storage tank low level alarm operable and set to alarm between 98,100 gallons and 100,850 gallons of water in the tank.

c. One residual heat removal pump and heat exchanger together with the associated piping and valves operable.

d. One recirculation pump together with its associated piping and valves operable.

2. If the Safety Injection and Residual Heat Removal Systems are not restored to meet the requirements of 3.3.A.1 within 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.

3. The reactor coolant system T_{avg} shall not exceed 350°F unless the following requirements are met:

a. The refueling water storage tank contains a minimum of 346,870 gallons of water at a boron concentration of at least 2000 ppm.

b. The boron injection tank contains 900 gallons of a boric acid solution of 11 1/2% to 13% by weight (20,000 ppm to 22,500 ppm of boron) at a temperature of at least 145°F. Two channels of heat tracing shall be operable for that portion of the flow path bounded by the boron injection tank inlet and outlet motor operated valves and the recirculation flow path to and from the boric acid tanks.

c. The four accumulators are pressurized between 600 and 700 psig and each contains a minimum of 800 ft^3 and a maximum of 815 ft^3 of water at a boron concentration of at least 2000 ppm. Accumulator isolation valves 894A, B, C, and D shall be open and their power supplies de-energized whenever the reactor coolant system pressure is above 1000 psig.

d. One pressure and one level transmitter shall be operating per accumulator.

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

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

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

h. Valves 856Band 856G in the Safety Injection discharge headers shall be closed and their power supplies de-energized.

i. Valve 1810 in the suction line of the high-head SI pumps and valves 882 and 744 in the suction and discharge lines, respectively, of the residual heat removal pumps shall be open and their power supplies de-energized.

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

k. The refueling water storage tank low level alarms are operable and set to alarm between 98,100 gallons and 100,850 gallons of water in the tank.

4. The requirements of 3.3.A.3 may be modified to allow any one of the following components to be inoperable at any one time:

a. The accumulators may be isolated during the performance of the reactor coolant system hydrostatic tests.

For the purpose of accumulator check valve leakage testing, one accumulator may be isolated at a time, for up to 8 hours, provided the reactor is in the hot shutdown condition.

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

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

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

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

f. One channel of heat tracing associated with the Boron Injection Tank and/or its recirculation lines 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.

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

5. If the Safety Injection and Residual Heat Removal Systems are not restored to meet the requirements of 3.3.A.3 within the time periods specified in 3.3.A.4; then:

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

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

   c. In either case, if the requirements of 3.3.A.3 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.

B. Containment Cooling and Iodine Removal Systems

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

   a. The spray additive tank contains a minimum of 4000 gallons of solution with a sodium hydroxide concentration of not less than 30% by weight.

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

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

a. Fan cooler unit 32, 34, or 35 or the flow path for fan coolant unit 32, 34, or 35 may be out of service for a period not to exceed 24 hours, provided both containment spray pumps are demonstrated to be operable.

OR

Fan cooler unit 31 or 33, or the flow path for fan cooler unit 31 or 33 may be out of service for a period not to exceed 7 days provided both containment spray pumps are demonstrated daily to be operable.

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

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

3. If the Containment Cooling and Iodine Removal are not restored to meet the requirements of 3.3.B.1 within the time periods specified in 3.3.B.2, then:

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

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

c. In either case, if the requirements of 3.3.B.1 are not satisfied within an additional 48 hours, the reactor shall be brought to the cold shtudown condition utilizing normal operating procedures. The shutdown shall start no later than the end of the 48 hour period.

C. Isolation Valve Seal Water System (IVSWS)

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

   a. The IVSWS shall be operable.

   b. The IVSW tank shall be maintained at a minimum pressure of 45 psig and contain a minimum of 144 gallons of water.

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

   a. Any one header of the IVSWS may be inoperable for a period not to exceed 4 consecutive days.

   b. Any valve required for the functioning of the system during and following accident conditions provided it is restored to an operable status within 4 days and all valves in the system that provide a duplicate function are demonstrated to be operable.

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

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

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

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

D. **Weld Channel and Penetration Pressurization System (WC & PPS)**

1. The reactor shall not be brought above the cold shutdown unless the electrical and mechanical penetrations and liner weld channels are continuously pressurized above 41 psig.

2. The requirements of 3.3.D.1 may be modified to allow any one header of the nitrogen or air pressurization system to be inoperable for a period not to exceed 4 consecutive days.

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

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

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

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

E. Component Cooling System

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

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

   b. Two auxiliary component cooling pumps, one per each recirculation pump, together with their associated piping and valves, are operable.

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

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

   a. One of the two operable component cooling pumps may be out of service, provided the pump is restored to operable status within 24 hours.

   b. Two auxiliary component cooling pumps serving the same recirculation pump may be out of service, provided at least one is restored to an operable status within 24 hours and at least one auxiliary component cooling pump serving the other recirculation pump is demonstrated to be operable.

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

3. If the Component Cooling System is not restored to meet the requirements of 3.3.E.1 within the time periods specified in 3.3.E.2, 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 3.3.E.1 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.

F. <u>Service Water System</u>

1. The reactor shall not be brought above cold shutdown unless three service water pumps on the designated essential header and a minimum of two service water pumps on the designated non-essential header together with their associated piping and valves are operable.

2. When the reactor is above cold shutdown and if the requirements of 3.3.F.1 cannot be met within twelve hours, the reactor shall be brought to the cold shutdown condition, starting no later than the end of the twelve hour period, utilizing normal operating procedures.

3. Isolation shall be maintained between the essential and non-essential headers at all times when above cold shutdown conditions except that for a period of eight hours the headers may be connected while another essential header is being placed in service as described in F.2, above.

G. Hydrogen Recombiner System

1. The reactor T_{avg} shall not exceed 350° unless the following requirements are met:

   a) Both hydrogen recombiner units together with their associated piping, valves, oxygen supply system and control system are operable.

   b) The containment atmosphere sampling system including the sampling pump, piping and valves are operable.

   c) Hydrogen and oxygen supplies shall not be connected to the hydrogen recombiner units except under conditions of an accident or those specified in 4.5.A.3.

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

a. One hydrogen recombiner unit or its associated flow path, or oxygen supply system or control system may be inoperable for a period not to exceed 15 days, provided the other recombiner unit is demonstrated to be operable.

b. One containment atmosphere sampling line may be inoperable for a period not to exceed 15 days, provided the other sampling lines are demonstrated to be operable.

c. The containment atmosphere sampling pump may be inoperable for a period not to exceed 15 days.

3. If the Hydrogen Recombiner System is not restored to meet the requirements of 3.2.G.1 within the time period specified in 3.3.G.2, then:

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

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

c. In either case, if the requirements of 3.3.G.1 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.

H. Control Room Ventilation System

1. The control room ventilation system shall be operable at all times when containment integrity is required as per Specificiation 3.6.

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

a. The control room ventilation system may be inoperable for a period not to exceed seven consecutive days. At the end of this seven day period if the mal-condition in the control room ventilation system has not been corrected, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If after an additional 48 hours the mal-condition still exists, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

Basis

The normal procedure for starting the reactor is, first, to heat the reactor coolant to near operating temperature, by running the reactor coolant pumps. The reactor is then made critical by withdrawing control rods and/or diluting boron in the coolant.[1] With this mode of startup, the energy stored in the reactor coolant during the approach to criticality is substantially equal to that during power operation, and, therefore, the minimum required engineered safeguards and auxiliary cooling systems are required to be operable.

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

The operable status of the various systems and components is demonstrated by periodic tests defined by Specification 4.5. A large fraction of

these tests will be performed while the reactor is operating in the power range. If a component is found to be inoperable, it will be possible in most cases to effect repairs and restore the system to full operability within a relatively short time. For a single component to be inoperable does not negate the ability of the system to perform its function,[2] but it reduces the redundancy provided in the reactor design and thereby limits the ability to tolerate additional equipment failures. To provide maximum assurance that the redundant component(s) will operate if required to do so, the redundant component(s) are to be tested prior to initiating repair of the inoperable component. If it develops that (a) the inoperable component is not repaired within the specified allowable time period, or (b) a second component in the same or related system is found to be inoperable, the reactor, if critical, will initially be brought to the hot shutdown condition utilizing normal operating procedures to provide for reduction of the decay heat from the fuel, and consequent reduction of cooling requirements after a postulated loss-of-coolant accident. This will also permit improved access for repairs in some cases. If the reactor was already subcritical, the reactor coolant system temperature and pressure will be maintained within the stated values in order to limit the amount of stored energy in the reactor coolant system. The stated tolerances provide a band for operator control. After a limited time in hot shutdown, if the malfunction(s) are not corrected, the reactor will be placed in the cold shutdown condition, utilizing normal shutdown and cooldown procedures. In the cold shutdown condition there is no possibility of an accident that would release fission products or damage the fuel elements.

The plant operating procedures require immediate action to effect repairs of an inoperable component, and, therefore, in most cases repairs will be completed in less than the specified allowable repair times. The limiting times to repair are based on two considerations:

1) Assuring with high reliability that the safeguard system will function properly if required to do so.

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

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

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

The limits for the Boron Injection Tank, Refueling Water Storage Tank, and the accumulators insure the required amount of water with the proper boron concentration for injection into the reactor coolant system following a loss-of-coolant accident is available. These limits are based on values used in the accident analysis.[9] [13]

The specified quantities of water for the RWST include unavailable water (4687 gals) in the tank bottom, inaccuracies (1406 gals) in the alarm setpoints, and minimum quantities required during injection (246,000 gals)[3] and recirculation phases (80,000 gals).[4] The minimum RWST (e.g., 346,870 gals) provides approximately 13,370 gallons margin.

The four accumulator isolation valves (894 A, B, C, D) are maintained in the open position when the reactor coolant pressure is above 1000 psig to assure flow passage from the accumulators will be available during the injection phase of a loss-of-coolant accident. Indication is also provided on the monitor light panel, should any of these valves not be in the full open position even with the valve operator de-energized. The 1000 psig limit is derived from the minimum pressure requirements of the accumulators combined with instrument error and an operational band and is based upon avoiding inadvertent injection into the reactor coolant system. The accumulator isolation valve motor operators are de-energized to prevent an extremely

unlikely spurious closure of these valves from occurring when accumulator core cooling flow is required. Valves 856 B and G are maintained in the closed position to prevent hot leg injection during the injection phase of a loss-of-coolant accident. As an additional assurance of preventing hot leg injection, these valve motor operators are de-energized to prevent spurious opening of these valves during the injection phase of a loss-of-coolant accident. Power will be restored to these valves at an appropriate time in accordance with plant operating procedures after a loss-of-coolant accident in order to establish hot leg recirculation.

Valves 1810, 882, and 744 are maintained in the open position to assure that flow passage from the refueling water storage tank will be available during the injection phase of a loss-of-coolant accident. As additional assurance of flow passage availability, these valve motor operators are de-energized to prevent an extremely unlikely spurious closure. This additional precaution is acceptable, since failure to maually re-establish power to close these valves following the injection phase is tolerable as a single failure.

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

With respect to the core cooling function, there is some functional redundancy for certain ranges of break sizes.[3] The measure of effectiveness of the Safety Injection System is the ability of the pumps and accumulators to keep the core flooded or to reflood the core rapidly where the core has been uncovered for postulated large area ruptures. The result of their performance is to sufficiently limit any increase in clad temperature below a value where emergency core cooling objectives are met.[13]

The containment cooling and iodine removal functions are provided by two independent systems: (a) fan-coolers plus charcoal filters and (b) containment spray with sodium hydroxide addition. During normal power operation , the five fan-coolers are required to remove heat lost from equipment and piping within containment at design conditions (with a cooling water temperature of 85°F).[4] In the event of a Design Basis Accident, any one of the following combinations will provide sufficient cooling to reduce containment pressure at a rate consistent with limiting off-site doses to acceptable values: (1) five fan-cooler units, (2) two containment spray pumps, (3) three fan-cooler units and one spray pump. Also in the event of a Design Basis Accident, three charcoal filters (and their associated recirculation fans) in operation, along with one containment spray pump and sodium hydroxide addition, will reduce airborne organic and molecular iodine activities sufficiently to limit off-site doses to acceptable values.[5] These constitute the minimum safeguards for iodine removal, and are capable of being operated on emergency power with one diesel generator inoperable.

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

Due to the distribution of the five fan cooler units and two containment spray pumps on the 480 volt buses, the closeness to which the combined equipment approaches minimum safeguards varies with which particular component is out of service. Accordingly, the allowable out of service periods vary according to which component is out of service. Under no conditions do the combined equipment degrade below minimum safeguards.

The four day out of service period for the Weld Channel and Penetration Pressurization System and the Isolation Valve Seal Water System is allowed because no credit has been taken for operation of these systems in the calculation of off-site accident doses should an accident occur. No other safeguards systems are dependent on operation of these systems.[11] The minimum pressure settings for the IVSWS and WC & PPS during operation assures effective performance of these systems for the maximum containment calculated peak accident pressure of 40.6 psig.[12]

The Component Cooling System is not required during the injection phase of a loss-of-coolant accident. The component cooling pumps are located in the Primary Auxiliary Building and are accessible for repair after a loss-of-coolant accident.[6] During the recirculation phase following a loss-of-coolant accident, only one of the three component cooling pumps is required for minimum safeguards.[7]

A total of six service water pumps are installed, only two of the set of three service water pumps on the header designated the essential header are required immediately following a postulated loss-of-coolant accident.[8] During the recirculation phase of the accident, two service water pumps on the non-essential header will be manually started to supply cooling water for one component cooling system heat exchanger, one control room air conditioner, and one diesel generator; the other component cooling system heat exchanger, the other control room air conditioner, the two other diesel generators and remaining safety related equipment are cooled by the essential service water header.[14]

Two full rated recombination systems are provided in order to control the hydrogen evolved in the containment following a loss-of-coolant accident. Either system is capable of preventing the hydrogen concentration from exceeding 2% by volume within the containment. Each of the systems is separate from the other and is provided with redundant features. Power supplies for the blowers and ignitors are separate, so that loss of one power supply will not affect the remaining system. Hydrogen gas is used as the externally supplied fuel. Oxygen gas is added to the containment atmosphere through a separate containment feed to prevent depletion of oxygen in the air below the concentration required for stable operation of the combustor (12%). The containment atmosphere

sampling system consists of a sample line which originates in each of the containment fan cooler units. The fan and sampling pump head together are sufficient to pump containment air in a loop from the fan cooler through a containment penetration to a sample vessel outside the containment. The design hydrogen concentration for operating the recombiner is established at 2% by volume. Conservative calculations indicate that the hydrogen content within the containment will not reach 2% by volume until 12 days after a loss-of-coolant accident.[10] There is, therefore, no need for immediate operation of the recombiner following an accident, and the quantity of hydrogen fuel stored at the site will be only for periodic testing of the recombiners.

Auxiliary Component Cooling Pumps are provided to deliver cooling water for the two Recirculation Pumps located inside the containment. Each recirculation pump is fed by two Auxiliary Component Cooling Pumps. A single Auxiliary Component Cooling Pump is capable of supplying the necessary cooling water required for a recirculation pump during the recirculation phase following a loss-of-coolant accident.

The control room ventilation system is designed to filter the control room atmosphere for intake air and/or for recirculation during control room isolation conditions. The control room system is designed to automatically start upon control room isolation and to maintain the control room pressure to the design positive pressure so that all leakage should be out leakage.

References

1) FSAR Section 9
2) FSAR Section 6.2
3) FSAR Section 6.2
4) FSAR Section 6.3
5) FSAR Section 14.3.5
6) FSAR Section 1.2
7) FSAR Section 8.2
8) FSAR Section 9.6.1
9) FSAR Section 14.3
10) FSAR Section 6.8
11) FSAR Section 6.5
12) Response to Question 14.6, FSAR Volume 7
13) FSAR Appendix 14C
14) Response to Question 9.35, FSAR Volume 7

3.4 STEAM AND POWER CONVERSION SYSTEM

Applicability

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

Objective

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

Specification

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

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

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

(3) A minimum of 360,000 gallons of water in the condensate storage tank.

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

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

(6) Two steam generators capable of performing their heat transfer function.

(7) City water system piping and valves directly associated with providing backup supply to the auxiliary feedwater pumps are operable.

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

C. The gross turbine-generator electrical output at all times shall be within the limitations of Figure 3.4-1 or Figure 3.4-2 for the applicable conditions of turbine overspeed setpoint, number of operable low pressure steam dump lines, and condenser backpressure as noted thereon.

Basis

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

The twenty main steam safety valves have a total combined rated capability of 15,108,000 lbs/hr. The total full power steam flow is 12,974,500 lbs/hr, therefore twenty (20) main steam safety valves will be able to relieve the total steam flow if necessary.

In the unlikely event of complete loss of electrical power to the station, decay heat removal would continue to be assured by the availability of either the steam-driven auxiliary feedwater pump or one of the two motor-driven auxiliary steam generator feedwater pumps, and steam discharge to the atmosphere via the main steam safety valves and atmospheric relief valves. One motor-driven auxiliary feedwater pump can supply sufficient

feedwater for removal of decay heat from the plant. The minimum amount of water in the condensate storage tank is the amount needed for 24 hours at hot shutdown. When the condensate storage supply is exhausted, city water will be used.

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

The limitations placed on turbine-generator electrical output due to conditions of turbine overspeed setpoint, number of operable steam dump lines, and condenser back pressure are established to assure that turbine overspeed (during conditions of loss of plant load) will be within the design overspeed value considered in the turbine missile analysis.[2] In the preparation of Figures 3.4-1 and 3.4-2, the specified number of operable L.P. steam dump lines is shown as one (1) greater than the minimum number required to act during a plant trip. The limitations on electrical output, as indicated in Figures 3.4-1 and 3.4-2, thus consider the required performance of the L.P. Steam Dump System in the event of a single failure for any given number of operable dump lines.

INDIAN POINT UNIT NO.3

Curve of Power Level versus Number of Operable Dump Lines with Parameters of Trip Set Point Required to Limit Maximum Overspeed to 132% Based on 1.0" Hg abs. Condenser Pressure

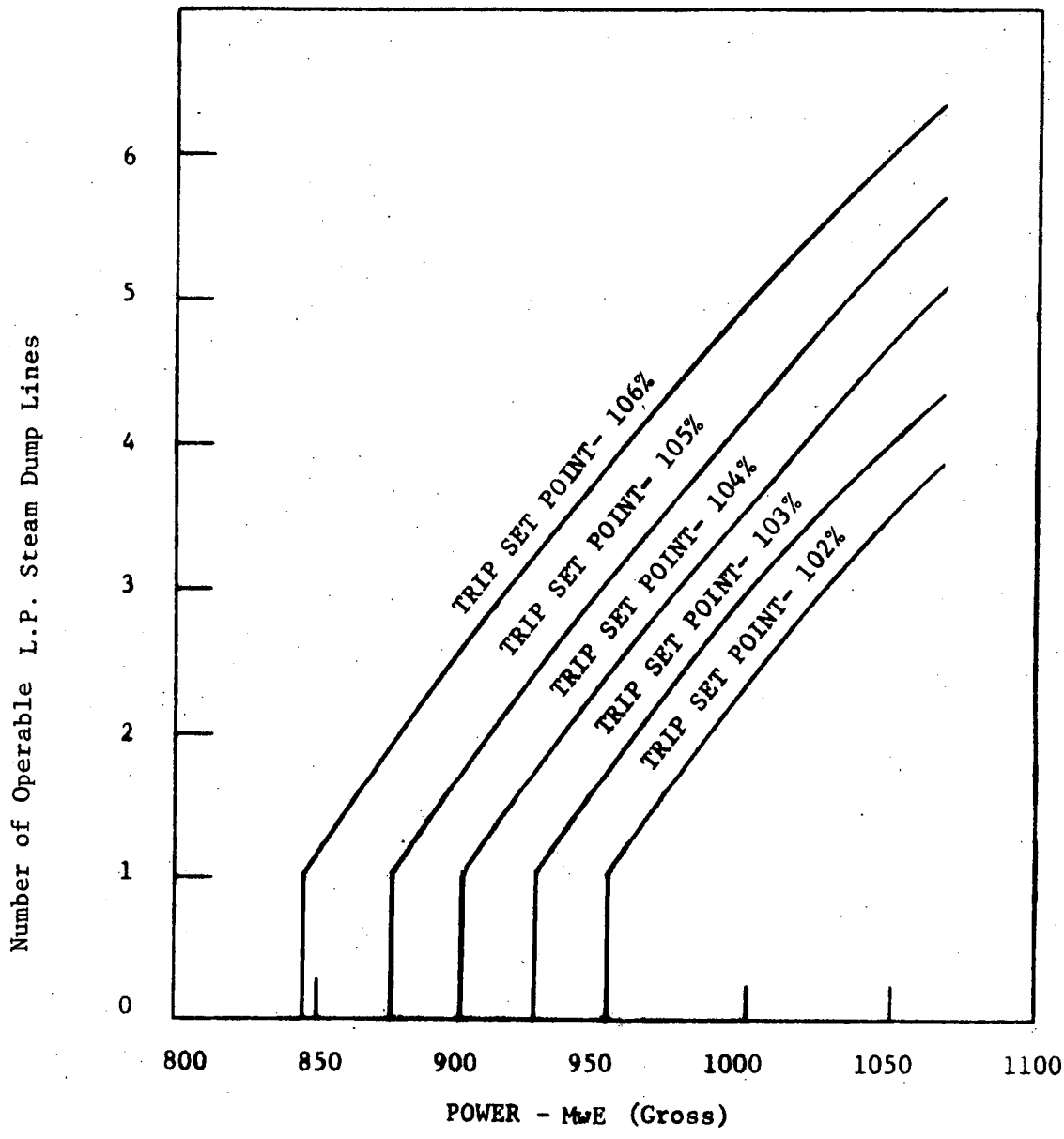


Figure 3.4-1 Gross Electrical Output
1.0 inch Hg Backpressure

INDIAN POINT UNIT NO.3

Curve of Power Level versus Number of Operable L.P. Dump Lines with Parameters of Trip Set Point Required to Limit Maximum Overspeed to 132% Based on 1.5" Hg abs. Condenser Pressure

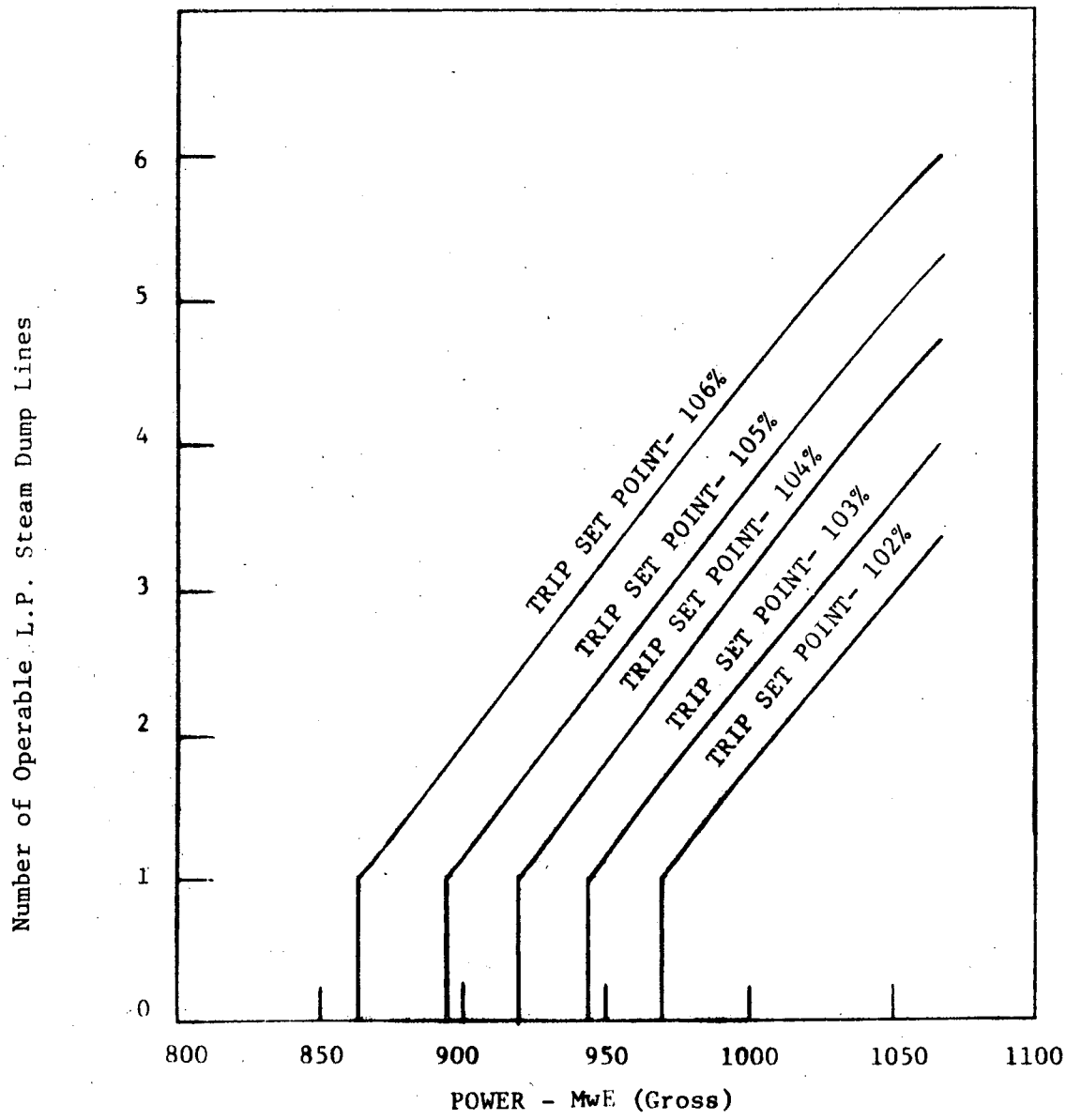


Figure 3.4-2 Gross Electrical Output 1.5 inch Hg Backpressure

3.5 INSTRUMENTATION SYSTEMS

Operational Safety Instrumentation

Applicability

Applies to plant instrumentation systems.

Objectives

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

Specification

3.5.1 When the plant is not in the cold shutdown condition, the Engineered Safety Features initiation instrumentation setting limits shall be as stated in Table 3.5-1.

3.5.2 For instrumentation testing or instrumentation channel failure, plant operation shall be permitted to continue in accordance with Tables 3.5-2 through 3.5-4. No more than one channel of a particular protection channel set shall be tested at the same time. By definition, an instrumentation channel failure shall not be regarded as a channel being tested.

3.5.3 In the event the number of in-service channels of a particular function is less than the minimum number of Operable Channels (Col. 3), or the Minimum Degree of Redundancy (Col. 4) cannot be achieved, operation shall be limited according to the requirement shown in Column 5 of Tables 3.5-2 through 3.5-4.

3.5.4 In the event of instrumentation channel failure permitted by specification 3.5.2, the Minimum Degree of Redundancy listed in Tables 3.5-2 through 3.5-4 may be reduced by one, but to not less than zero, and the Minimum Number of Operable Channels listed in these tables may be reduced by one, but not to less than one (except as noted in Table 3.5-3) for a period of 8 hours while instrument channels are tested. The failed channel may be blocked to prevent an unnecessary reactor trip during this time. In the case of three loop operation, the out-of-service channel is permitted to be bypassed during the test period.

3.5.5 The coincident low pressurizer pressure/pressurizer level safety injection trip shall be unblocked when the pressurizer pressure is $\geq$ 2000 psig.

3.5.6 At least one source range and one intermediate range nuclear instrument channel shall be operable prior to reactor start-up.

3.5.7 When the reactor is not in the cold shutdown condition, the instrumentation requirements as stated in Table 3.5-5 shall be met.

Basis

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

Safety Injection System Actuation

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

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

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

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

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

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

Containment Spray

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

Containment spray is designed to be actuated at a higher containment pressure (∿ 50% of containment design pressure) than the SIS (∿ 10% of containment design pressure). Since spurious actuation of containment spray is to be avoided, it is automatically initiated only on coincidence of Hi-Hi Level containment pressure sensed by both sets of two-out-of-three containment pressure signals and coincidence with the S.I. Signal.

Steam Line Isolation

Steam line isolation signals are initiated by the Engineered Safety Features closing all steam line stop valves. In the event of a steam line break, this action prevents continuous, uncontrolled steam release from more than one steam generator by isolating the steam lines on high containment pressure (Hi-Hi Level) or high steam line flow. Protection is afforded for breaks inside or outside the containment even when it is assumed that there is a single failure in the steam line isolation system.

Feedwater Line Isolation

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

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

Setting Limits

1. The Hi-Level containment pressure limit is set at about 10% of containment design pressure. Initiation of Safety Injection protects against loss of coolant[2] or steam line break[3] accidents as discussed in the safety analysis.

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

3. The pressurizer low pressure-low level limit is set substantially below system operating pressure limits. However, it is sufficiently high to protect against a loss of coolant accident as shown in the safety analysis[2]. The trip is bypassed below 2000 psig to prevent inadvertent actuation of the Engineered Safeguards when the reactor is shutdown.

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

5. The high steam line flow measurement ΔP limit is set at approximately 40% of the full steam flow from no load to 20% load. Between 20% and 100% (full) load, the trip setpoint for the flow measurement ΔP is ramped linearly with respect to first stage turbine pressure from 40% of the full steam flow to 110% of the full steam flow. These setpoints will initiate safety injection in the case of a large steam line break accident. Coincident low Tavg setting limit for SIS and steam line isolation initiation is set below the hot shutdown value. The coincident steam line pressure setting limit is set below the full load operating pressure. The safety analysis shows that these settings provide protection in the event of a large steam line break.[3]

Instrument Operating Conditions

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

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

In the event that either the specified Minimum Number of Operable Channels or the Minimum Degree of Redundancy cannot be met, the reactor and the remainder of the plant is placed, utilizing normal operating procedures, in that condition consistent with the loss of protection.

The source range and the intermediate range nuclear instrumentation and the turbine and steam-feedwater flow mismatch trip functions are not required to be operable since they were not used in the transient and safety analysis (FSAR Section 14).

References:

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

TABLE 3.5-1

ENGINEERED SAFETY FEATURES INITIATION INSTRUMENT SETTING LIMITS

No.	FUNCTIONAL UNIT	CHANNEL	SETTING LIMIT
1.	High Containment Pressure (Hi Level)	Safety Injection	$\leq$ 3.5 psig
2.	High Containment Pressure (Hi-Hi Level)	a. Containment Spray b. Steam Line Isolation	$\leq$ 23 psig
3.	Pressurizer Low Pressure and Low Level	Safety Injection	$\geq$ 1700 psig $\geq$ 5 percent instrument span
4.	High Differential Pressure Between Steam Lines	Safety Injection	$\leq$ 150 psi
5.	High Steam Flow in 2/4 Steam Lines Coincident with Low T_{avg} or Low Steam Line Pressure	a. Safety Injection b. Steam Line Isolation	$\leq$ 40% of full steam flow at zero load $\leq$ 40% of full steam flow at 20% load $\leq$ 110% of full steam flow at full load $\geq$ 540°F T_{avg} $\geq$ 600 psig steam line pressure

TABLE 3.5-2 (Sheet 1 of 2)

REACTOR TRIP INSTRUMENTATION LIMITING OPERATING CONDITIONS

NO.	FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MIN. NUMBER OF OPERABLE CHANNELS	4 MIN. DEGREE OF REDUNDNACY	5 OPERATOR ACTION IF CONDITIONS OF COL. 3 OR 4 CANNOT BE MET*
1.	Manual Reactor Yrip	2	1	1	0	Maintain hot shutdown
2.	Nuclear Flux Power Range	4	2	3	2	Maintain hot shutdown
		4	2	2	1	For zero power physics tests only
3.	Overtemperature ΔT	4	2	3	2	Maintain hot shutdown
4.	Overpower ΔT	4	2	3	2	Maintain hot shutdown
5.	Low Pressurizer Pressure	4	2	3	2	Maintain hot shutdown
6.	Hi Pressurizer Pressure	3	2	2	1	Maintain hot shutdown
7.	Pressurizer-Hi Water Level	3	2	2	1	Maintain hot shutdown
8.	Low Flow One Loop (Power ≥ P-8)	3/loop	2/loop (any loop)	2/operable loop	1/operable loop	Maintain hot shutdown
	Low Flow Two Loops (Power < P-8 and ≥ P-10)	3/loop	2/loop (any two loops)	2/operable loop	1/operable loop	Maintain hot shutdown

TABLE 3.5-2 (Sheet 2 of 2)

		1	2	3	4	5
9.	Lo Lo Steam Generator Water Level	3/loop	2/loop	2/loop	1/loop	Maintain hot shutdown
10.	Undervoltage 6.9 KV Bus	1/bus	2	3	2	Maintain hot shutdown
11.	Low Frequency 6.9 KV Bus**	1/bus	2	3	2	Maintain hot shutdown
12.	Turbine trip (electrical over-speed protection)	3	2	2	1	Turbine shutdown (turbine stop valves closed)

* Maintain hot shutdown means maintain or proceed to hot shutdown within 4 hours using normal operating procedures, if the unacceptable condition arises during operation.

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

TABLE 3.5-3 (Sheet 1 of 2)

INSTRUMENTATION OPERATING CONDITION FOR ENGINEERED SAFETY FEATURES

NO.	FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MIN. NUMBER OF OPERABLE CHANNELS	4 MIN. DEGREE OF REDUNDANCY	5 OPERATOR ACTION IF CONDITIONS OF COL. 3 OR 4 CANNOT BE MET*****
1.	SAFETY INJECTION					
	a. Manual	2	1	1	0	Cold Shutdown
	b. High Containment Pressure (Hi Level)	3	2	2	1	Cold Shutdown
	c. High Differential Pressure Between Steam Lines	3/steam line	2/steam line	2/steam line	1/steam line	Cold Shutdown
	d. Pressurizer Low Pressure and Low Level*	3**	1**	2**	1	Cold Shutdown
	e. High Steam Flow in 2/4 Steam Lines Coincident with Low Tavg or Low Steam Line Pressure	2/steam line	1/2 in any 2 steam lines	2 channels in each of 3 steam lines	2	Cold Shutdown or main steam isolation valves closed
		4 Tavg Signals	2	3	2	
		4 Pressure Signals	2	3	2	
	f. Pressurizer Low Pressure and Low Level (Automatic Unblock)	3	2	2****	1****	Cold Shutdown

TABLE 3.5-3 (Sheet 2 of 2)

	1	2	3	4	5
2. CONTAINMENT SPRAY					
a. Manual	2	2	2	0***	Cold Shutdown
b. High Containment Pressure (Hi Hi Level)	2 sets of 3	2 of 3 in each set	2 per set	1/set	Cold Shutdown

* Permissible to bypass if reactor coolant pressure less than 2000 psig.

** Each channel has two separate signals.

*** Must actuate 2 switches simultaneously.

**** The Minimum Number of Operable Channels and the Minimum Degree of Redundancy may be reduced to zero if the SI bypass is in the unblocked position.

***** If the condition of Column 3 or 4 cannot be met, the reactor shall be placed in the hot shutdown condition, utilizing normal operating procedures, within 4 hours of the occurrence. If the conditions are not met within 24 hours of the occurrence, the reactor shall be placed in the cold shutdown condition, or the alternate condition, if applicable, within an additional 24 hours.

TABLE 3.5-4 (Sheet 1 of 2)

INSTRUMENT OPERATING CONDITIONS FOR ISOLATION FUNCTIONS

NO.	FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MIN. OPERABLE CHANNELS	4 MIN. DEGREE OF REDUN-ANCY	5 OPERATOR ACTION IF CONDITIONS OF COLUMN 3 OR 4 CANNOT BE MET*
1.	CONTAINMENT ISOLATION					
	a. Automatic Safety Injection (Phase A)	See Item No. 1(b) of Table 3.5-3				Cold Shutdown
	b. Containment Pressure (Phase B)	See Item No. 2(b) of Table 3.5-3				Cold Shutdown
	c. Manual					
	Phase A	2	1	1	0	Cold Shutdown
	Phase B	See Item 2(a) of Table 3.5-3				Cold Shutdown
2.	STEAM LINE ISOLATION					
	a. High Steam Flow in 2/4 Steam Lines Coincident with Low T_{avg} or Low Steam Line Pressure	See Item No. 1(e) of Table 3.5-3				Cold Shutdown or Main Steam Isolation Valves Closed
	b. High Containment Pressure (Hi Hi Level)	See Item No. 2(b) of Table 3.5-3				Cold Shutdown or Main Steam Isolation Valves Closed**
	c. Manual	1/loop	1/loop	1/loop	0	Cold Shutdown or Main Steam Isolation Valves Closed

TABLE 3.5-4 (Sheet 2 of 2)

	1	2	3	4	5
3. FEEDWATER LINE ISOLATION					
a. Safety Injection	See Item No. 1 of Table 3.5-3				

* If the conditions of Columns 3 or 4 cannot be met, the reactor shall be placed in the hot shutdown condition, utilizing normal operating procedures, within 4 hours of the occurrence. If the conditions are not met within 24 hours of the occurrence, the reactor shall be placed in the cold shutdown condition, or the alternate condition if applicable, within an additional 24 hours.

** Main steam isolation valves may be closed in lieu of going to cold shutdown if the circuitry associated with closing the valves is the only portion inoperable.

TABLE 3.5-5 (Sheet 1 of 2)

TABLE OF INDICATORS AND/OR RECORDERS AVAILABLE TO THE OPERATOR

	PARAMETER	1 NO. OF CHANNELS AVAILABLE	2 MIN. NO. OF CHANNELS REQUIRED**	3 INDICATOR/ RECORDER**
1)	Containment Pressure	6	1	Indicator
2)	Refueling Water Storage Tank Level	2	1	Indicator
3)	Steam Generator Water Level (Narrow Range)	3/steam generator	*	Indicator
4)	Steam Generator Water Level (Wide Range)	1/steam generator	*	Recorder
5)	Steam Line Pressure	3/steam line	1/steam line	Indicator
6)	Pressurizer Water Level	3	1	Indicator/One Channel is recorded
7)	RHR Recirculation Flow	4	3	Indicator
8)	Reactor Coolant System Pressure (Wide Range)	1	1	Recorder
9)	Cold Leg Temperature (Tc) (Wide Range)	4	1	Recorder
10)	Hot Leg Temperature (Th) (Wide Range)	4	1	Recorder
11)	Containment Sump Level	2	1	Indicator
12)	Recirculation Sump Level	2	1	Indicator
13)	Temperature Sensors in Penetration Area of Primary Auxiliary Building	3	1	Alarm
14)	Temperature Sensors in Auxiliary Boiler Feedwater Pump Building	2	1	Alarm

TABLE 3.5-5 (Sheet of 2 of 2)

	1	2	3
15) Level Sensors in Lower Level of Turbine Building	2	1	Alarm

* One level channel per steam generator (either wide range or narrow range) with at least two wide range channels.

** Columns 2 and 3 may be modified to allow the instrument channels to be inoperable for up to 7 days and/or the recorders to be inoperable for up to 14 days.

If the minimum number of channels required are not restored to meet the above requirements within the time periods specified, 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 Columns 2 and 3 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.

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 1.10) 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 $\geq 10\% \ \frac{\Delta k}{k}$.

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. Internal Pressure

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. Containment Temperature

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

BASIS

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

The shutdown margins are selected based on the type of activities that are being carried out. The 10% $\Delta k/k$ shutdown margin when the head is off 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]

Table 3.6-1 lists non-automatic valves that are designated as part of the containment isolation function [4]. During periods of normal plant operations requiring containment integrity, valves on this Table will be open either continuously or intermittently depending on requirements of the particular protection, safeguards or essential service systems. Those valves to be open intermittently are under administrative control and are open only as long as necessary to perform their intended function. In all cases, however, the valves listed in Table 3.6-1 are closed during the post accident period in accordance with plant procedures and consistant with requirements of the related protection, safeguards, or essential service systems.

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

(4) FSAR - Section 5.2

TABLE 3.6-1

NON-AUTOMATIC CONTAINMENT ISOLATION VALVES
OPEN CONTINUOUSLY OR INTERMITTENTLY FOR PLANT OPERATION

550	752F	SWN-41	SWN-44
744	753F	SWN-43	SWN-51
1870	752J		
743	753J	SWN-41	
732	891A	SWN-43	SWN-71
885A	891B		SWN-71
885B	891C	SWN-41	SWN-71
205	891D	SWN-43	SWN-71
226	863		SWN-71
227	878A	SWN-41	
	878B		
250A	PCV-1111	SWN-43	UH-37
241A	PCV-1111		UH-38
250B	1814A	SWN-41	1882A
241B	1814B	SWN-43	1875A
250C	1814C		1875B
241C	859A	SWN-44	1876A
250D	859C	SWN-51	1876B
241D	1833A	SWN-44	PS-7
869A	1833B	SWN-51	PS-8
869B	SA-24	SWN-44	PS-9
851A	SA-24	SWN-51	PS-10
850A	580A	SWN-44	888A
1610	580B	SWN-51	888B
990A	958		1890A
990B	959		1890B
	990C		1890C
			1890D
			1890E
			1890F
			1890G
			1890H
			1890J

3.7 AUXILIARY ELECTRICAL SYSTEMS

Applicability

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

Objective

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

Specification

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

1. Two physically independent transmission circuits to Buchanan Substation capable of supplying engineered safeguards loads.

2. 6.9 KV buses 5 and 6 energized from either 138 KV feeder 95331 or 95332.

3. Either 13.8 KV feeder 13W92 or 13W93 and its associated 13.8/6.9 KV transformer available to supply 6.9 KV power.

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

5. Three diesel generators operable with a minimum onsite supply of 5676 gallons of fuel available in each of the three individual underground storage tanks and 26,300 gallons of fuel compatible for operation with the diesels available onsite other than the

underground storage tanks or at the Buchanan substation. This 26,300 gallon reserve is for Indian Point Unit No. 3 usage only and is in addition to the fuel requirements for other nuclear units on the site.

6. Three batteries plus three chargers and the D. C. distribution systems operable.

B. The requirements of 3.7.A may be modified to allow any one of the following power supplies to be inoperable at any one time:

1. One diesel or any diesel fuel oil system or a diesel and its associated fuel oil system may be inoperable for up to 7 days provided the 138 KV and the 13.8 KV sources of offsite power are available and the remaining diesel generators are tested daily to ensure operability and the engineered safety features associated with these diesel generator buses are operable.

2. The 138 KV or the 13.8 KV sources of power may be inoperable for 48 hours provided the three diesel generators are operable. This operation may be extended beyond 48 hours provided the failure is reported to the NRC within the 48 hour period with an outline of the plans for restoration of offsite power and NRC approval is granted.

3. One battery may be inoperable for 24 hours provided the other batteries and the three battery chargers remain operable with one battery charger carrying the D. C. load of the failed battery supply system.

C. If the electrical distribution system is not restored to meet the requirements of 3.7.A within the time periods specified in 3.7.B, 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 3.7.A 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.

D. The requirements of Specification 3.7.A.1 may be modified during an emergency system-wide blackout condition as follows:

Two of the three 13.8 KV feeders (13W92, 13W93 and/or 13W94) to the Buchanan Substation 138 KV buses operable with at least 37 MW power from any combination of gas turbines (nameplate rating at 80°F) at the Buchanan Substation and onsite available for exclusive use on Indian Point Unit No. 3.

Basis

The electrical system equipment is arranged so that no single contingency can inactivate enough safeguards equipment to jeopardize the plant safety. The 480-volt equipment is arranged on 4 buses. The 6900-volt equipment is supplied from 6 buses.

The Buchanan Substation has both 345 KV and 138 KV transmission circuits which are capable of supplying startup, normal operation, shutdown and/or engineered safeguards loads.

The 138 KV supplies or the gas turbines are capable of providing sufficient power for plant startup. Power via the station auxiliary transformer can supply all the required plant auxiliaries during normal operation, if required.

In addition to the unit transformer, four separate sources supply station service power to the plant.[1]

The plant auxiliary equipment is arranged electrically so that multiple items receive their power from different buses. Redundant valves are individually supplied from separate motor control centers.

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

Two diesel generators have sufficient capacity to start and run within design load the minimum required engineered safeguards equipment.(1) The minimum onsite underground stored diesel fuel oil inventory is maintained at all times to assure the operation of two diesels carrying the minimum required engineered safeguards equipment load for at least 48 hours.(2) Additional fuel oil suitable for use in the diesel generators will be stored either on site or at the Buchanan Substation. The minimum storage of 26,300 gallons of additional fuel oil will assure continuous operation of two diesels at the minimum engineered safeguards load for a total of 7 days. A truck with hosing connections compatible with the underground diesel fuel oil storage tanks is available for transferal of diesel oil from storage areas either on site or at the Buchanan Substation. Commercial oil supplies and trucking facilities are also available.

Periodic diesel outages will be necessary to perform the corrective maintenance required as a result of previous tests or operations and the preventive maintenance recommended by the manufacturer.

One battery charger shall be in service on each battery so that the batteries will always be at full charge in anticipation of a loss-of-AC power incident. This insures that adequate D.C. power will be available for starting the emergency generators and other emergency uses.

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

Any two of three diesel generators, the station auxiliary transformer or the separate 13.8 to 6.9 KV transformer are each capable of supplying the minimum safeguards loads, and therefore provide separate sources of power immediately available for operation of these loads. Thus the power supply system meets the single failure criteria required of safety systems. To provide maximum assurance that the redundant or alternate power supplies will operate if required to do so, the redundant or alternate power supplies are verified operable prior to initiating repair of the inoperable power supply. If it develops that (a) the inoperable power supply is not repaired within the specified allowable time period, or (b) a second power supply in the same or related category is found to be inoperable, the reactor, if critical, will initially be brought to the hot shutdown condition utilizing normal operating procedures to provide for reduction of the decay heat from the fuel, and consequent reduction of cooling requirements after a postulated loss-of-coolant accident. If the reactor was already subcritical, the reactor coolant system temperature and pressure will be maintained within the stated values in order to limit the amount of stored energy in the Reactor Coolant System. The stated tolerances provide a band for operator control. After a limited time in hot shutdown, if the malfunction(s) are not corrected, the reactor will be brought to the cold shutdown condition, utilizing normal shutdown and cool-down procedures. In the cold shutdown condition there is no possibility of an accident that would release fission products or damage the fuel elements.

Conditions of a system-wide blackout could result in a unit trip. Since normal off-site power supplies as required in Specification 3.7.A.1 are not available for startup, it is necessary to be able to black start the unit with gas turbines providing the incoming power supplies as a first step in restoring the system to an operable status and restoring power to customers for essential services. Specification 3.7.C provides for startup using 37 MW's of gas turbine power (nameplate rating at 80°F) which is sufficient to carry out a normal plant startup. A system-wide blackout is deemed to exist when the majority of Con Edison electric generating facilities are shutdown due to an electrical disturbance and the remainder are incapable of supplying the system therefore necessitating major load shedding.

Reference

1) FSAR - Section 8.2.1

2) FSAR - Section 8.2.3

3.8 REFUELING, FUEL HANDLING AND STORAGE

Applicability

Applies to operating limitations during refueling, fuel handling, and storage operations.

Objective

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

Specification

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

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

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

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

4. At least one residual heat removal pump and heat exchanger shall be 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 initiate isolation, shall be tested and verified to be operable within 100 hours prior to refueling operations.

8. No movement of fuel in the reactor shall be made until the reactor has been subcritical for at least 100 hours.

9. Whenever movement of irradiated fuel is being made, the minimum water level in the area of movement shall be maintained 23 feet over the top of irradiated fuel assemblies seated within the reactor pressure vessel.

10. Hoists or cranes utilized in handling irradiated 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.

11. The fuel storage building emergency ventilation system shall be operable whenever irradiated fuel is being handled within the fuel storage building. The emergency ventilation system may be inoperable when irradiated fuel is in the fuel storage building, provided irradiated fuel is not being handled and neither the spent fuel cask nor the cask crane are moved over the spent fuel pit during the period of inoperability.

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.

C. During fuel handling and storage operations, the following conditions shall be met.

1. Radiation levels in the spent fuel storage area shall be monitored continuously whenever there is irradiated fuel stored therein. If the monitor is inoperable, a portable monitor may be used.

2. The spent fuel cask shall not be moved over any region of the spent fuel pit which contains irradiated fuel. Additionally, if the spent fuel pit contains irradiated fuel, the spent fuel cask shall not be moved over any region of the spent fuel pit unless the irradiated fuel stored therein has been in a subcritical condition for at least 90 days.

3. During periods of spent fuel cask or fuel storage building cask crane movement over the spent fuel pit, when the pit contains irradiated fuel, the pit shall be filled with borated water at a concentration of > 1000 ppm.

4. Whenever movement of irradiated fuel in the spent fuel pit is being made, the minimum water level in the area of movement shall be maintained 23 feet over the top of irradiated fuel assemblies seated in the storage rack.

5. Hoists or cranes utilized in handling irradiated 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 fuel handling operation. A thorough visual inspection of the hoists or cranes shall be made after the dead-load test and prior to fuel handling.

6. The fuel storage building emergency ventilation system shall be operable whenever irradiated fuel is being handled within the fuel storage building. The emergency ventilation system may be inoperable when irradiated fuel is in the fuel storage building, provided irradiated fuel is not being handled and neither the spent fuel cask nor the cask crane are moved over the spent fuel pit during the period of inoperability.

Basis

The equipment and general procedures to be utilized during refueling, fuel handling, and storage 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, fuel handling, or storage 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 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 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 342,000 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. The requirement for direct communications 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 100-hour decay time following the subcritical condition and the 23 feet of water above the top of the irradiated fuel assemblies 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.

The minimum spent fuel pit boron concentration and the 90-day restriction on the movement of the spent fuel cask to allow the irradiated fuel to decay were specified in order to minimize the consequences of an unlikely sideways cask drop.

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 a 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 irradiated 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

3.9 RADIOACTIVE MATERIALS MANAGEMENT

Applicability

Applies to the handling and use of sealed special nuclear, source and by-product material.

Objective

To assure that leakage from by-product, source, and special nuclear radioactive material sources does not exceed allowable limits.

Specification

A. Tests for leakage and/or contamination shall be performed as follows:

1. Each sealed source, except startup sources subject to core flux, containing radioactive material, other than Hydrogen-3, with a half life greater than thirty days and in any form other than gas shall be tested for leakage and/or contamination at intervals not to exceed six months.

2. The periodic leak test required does not apply to sealed sources that are stored and not being used. The sources excepted from this test shall be tested for leakage prior to any use or transfer to another user unless they have been leak tested within six months prior to the date or use or transfer. In the absence of a certificate from a transferor indicating that a test has been made within six months prior to the transfer, sealed sources shall not be put into use until tested.

3. Startup sources shall be leak tested prior to being subjected to core flux and following repair or maintenance to the source.

B. Sealed sources are exempt from Specification 3.9.A when the source contains 100 microcuries or less of beta and/or gamma emitting material or 5 microcuries or less of alpha emitting material.

C. The leakage test shall be capable of detecting the presence of 0.005 microcurie of radioactive material on the test sample. If the test reveals the presence of 0.005 microcurie or more of removable contamination, the sealed source shall immediately be withdrawn from use and either decontaminated and repaired, or be disposed of in accordance with Commission regulations.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Applicability:

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

Objectives:

To ensure:

1. Core subcriticality after reactor trip,
2. Acceptable core power distribution during power operation in order to maintain fuel integrity in normal operation and transients associated with faults of moderate frequency, supplemented by automatic protection and by administrative procedures, and to maintain the design basis initial conditions for limiting faults, and
3. Limit potential reactivity insertions caused by hypothetical control rod ejection.

Specifications:

3.10.1 Shutdown Reactivity

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

3.10.2 Power Distribution Limits

3.10.2.1 At all times, except during low power physics tests, the hot channel factors defined in the basis must meet the following limits:

$$F_Q(Z) \leq (2.32/P) \times K(Z) \text{ for } P > 0.5$$
$$F_Q(Z) \leq (4.64) \times K(Z) \text{ for } P \leq 0.5$$
$$F^N_{\Delta H} \leq 1.55\ [1 + 0.2\ (1-P)]$$

where P is the fraction of full power at which the core is operating. K(Z) is the fraction given in Figure 3.10-2 and Z is the core height location of F_Q.

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

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

3.10.2.2.2 The measurement of enthalpy rise hot channel factor, $F_{\Delta H}^N$, shall be increased by four percent to account for measurement error. If either measured hot channel factor exceeds its limit specified under Item 3.10.2.1, the reactor power and high neutron flux trip setpoint shall be reduced so as not to exceed a fraction of rated power equal to the ratio of the F_Q or $F_{\Delta H}^N$ limit to measured value, whichever is less. If subsequent in-core mapping cannot, within a 24-hour period, demonstrate that the hot channel factors are met, the reactor shall be brought to a hot shutdown condition with return to power authorized only for the purpose of physics testing.

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

3.10.2.4 Except during physics tests, during excore calibration procedures and except as modified by Items 3.10.2.5 through 3.10.2.7 below, the indicated axial flux difference of all but one operable excore channel shall be maintained within a $\pm$ 5% band about the target flux difference.

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

3.10.2.5.1 If the indicated axial flux difference of more than one operable excore channel deviates from its target band, either such deviation shall be immediately eliminated or the reactor power shall be reduced to a level no greater than 90 percent of rated power.

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

3.10.2.6.1 The indicated axial flux difference may deviate from its ± 5% target band for a maximum of one hour (cumulative) in any 24 hour period provided the flux difference does not exceed an envelope bounded by -11 percent and +11 percent at 90% power and increasing by -1 percent and +1 percent for each 2 percent of rated power below 90% power. A two hour deviation is permissible during tests performed as part of the augmented startup program.[1]

3.10.2.6.2 If Item 3.10.2.6.1 is violated by more than one operable excore channel, then the reactor power shall be reduced to no greater than 50% power and the high neutron flux setpoint reduced to no greater than 55 percent of rated values.

3.10.2.6.3 A power increase to a level greater than 90 percent of rated power is contingent upon the indicated axial flux difference of all but one operable excore channel being within their target band.

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

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

3.10.2.7.2 A power increase to a level greater than 50 percent of rated power is contingent upon the indicated axial flux difference of all but one operable excore channel not being outside their target bands for more than two hours (cumulative) out of the preceding 24-hour period. One-half the time the indicated axial flux difference is out of its target band up to 50% of rated power is to be counted as contributing to the one-hour cumulative (two-hour cumulative during augmented startup tests)[1] maximum the flux difference may deviate from its target band of a power level ≤ 90% of rated power.

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

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

3.10.3 Quadrant Power Tilt Limits

3.10.3.1 Whenever the indicated quadrant power tilt ratio exceeds 1.02, except for physics tests, within two hours the tilt condition shall be eliminated or the following actions shall be taken:

a) Restrict core power level and reset the power range high flux setpoint two percent of rated value for every percent of indicated power tilt ratio exceeding 1.0, and

b) If the tilt condition is not eliminated after 24 hours, the power range nuclear instrumentation setpoint shall be reset to 55% of allowed power. Subsequent reactor operation is permitted up to 50% for the purpose of measurement, testing and corrective action.

3.10.3.2 Except for physics tests, if the indicated quadrant power tilt ratio exceeds 1.09 and there is simultaneous indication of a misaligned control rod, restrict core power level 2% of rated value for every percent of indicated power tilt ratio exceeding 1.0 and realign the rod within two hours. If the rod is not realigned within two hours or if there is no simultaneous indication of a misaligned rod, the reactor shall be brought to the hot shutdown condition within 4 hours. If the reactor is shut down, subsequent testing up to 50% of rated power shall be permitted to determine the cause of the tilt.

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

3.10.3.4 The tilt deviation alarm shall be set to annunciate whenever the excore tilt ratio exceeds 1.02. If one or both of the quadrant power tilt monitors is inoperable, individual upper and lower excore detector calibrated outputs shall be logged once per shift and after a load change greater than 10 percent of rated power.

3.10.4 Rod Insertion Limits

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

3.10.4.2 When the reactor is critical, the control banks shall be limited in physical insertion to the insertion limits shown in Figure 3.10-4 or Figure 3.10-5.

3.10.4.3 Control bank insertion shall be further restricted if:

a. The measured control rod worth of all rods, less the worth of the most reactive rod (worst case stuck rod), is less than the reactivity required to provide the design value of available shutdown,
b. A rod is inoperable (Specification 3.10.7).

3.10.4.4 Full length control rod insertion limits do not apply during physics tests or during periodic exercise of individual rods. However, the shutdown margin indicated in Figure 3.10-1 must be maintained except for the low power physics test to measure control rod worth and shutdown margin. For this test, the reactor may be critical with all but one full length control rod inserted and part length rods fully withdrawn.

3.10.4.5 Part length rods shall not be permitted in the core except for low power physics tests and for axial offset calibration tests performed below 75% of rated power. Part length rods shall be limited in physical insertion to the insertion limits shown in Figure 3.10-3.

3.10.5 Rod Misalignment Limitations

3.10.5.1 If an indicated full length or part length rod cluster control assembly is misaligned from its bank demand position by more than 13 steps, then realign the rod or determine the core peaking factors within 2 hours and apply Specification 3.10.2.

3.10.5.2 If the requirements of Specification 3.10.3 are determined not to apply and the core peaking factors have not been determined within two hours and the rod remains misaligned, the high reactor flux setpoint shall be reduced to 85% of its rated value.

3.10.5.3 If the misaligned rod cluster control is not realigned within 8 hours, the rod shall be declared inoperable.

3.10.6 Inoperable Rod Position Indicator Channels

3.10.6.1 If a rod position indicator channel is out of service then:

a. For operation between 50 percent and 100 percent of rating, the position of the rod cluster control shall be checked indirectly by core instrumentation (excore detectors and/or movable incore detectors) every shift, or subsequent to rod motion exceeding 24 steps, whichever occurs first.

b. During operation below 50 percent of rating, no special monitoring is required.

3.10.6.2 Not more than one rod position indicator channel per group nor two rod position indicator channels per bank shall be permitted to be inoperable at any time.

3.10.6.3 If a full length or part length rod having a rod position indicator channel out of service, is found to be misaligned from 3.10.6.1a, above, then Specification 3.10.5 will be applied.

3.10.7 Inoperable Rod Limitations

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

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

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

3.10.8 Rod Drop Time

At operating temperature and full flow, the drop time of each full length rod cluster control shall be no greater than 1.8 seconds from loss of stationary gripper coil voltage to dashpot entry.

3.10.9 Rod Position Monitor

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

3.10.10 Notification

Any event requiring plant shutdown or trip setpoint reduction because of Specification 3.10 shall be reported to the Nuclear Regulatory Commission within 30 days.

BASIS

Design criteria have been chosen for normal operations, operational transients and those events analyzed in FSAR Section 14.1 which are consistent with the fuel integrity analyses. These relate to fission gas release, pellet temperature and cladding mechanical properties. Also, the minimum DNBR in the core must not be less than 1.30 in normal operation or in short term transients.

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

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

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

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

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

An upper bound envelope of 2.32 times the normalized peaking factor axial dependence of Figure 3.10-2 has been determined from extensive analyses considering all operating maneuvers consistent with the technical specifications on power distribution control as given in Section 3.10. The results of the loss of coolant accident analyses based on this upper bound normalized envelope of Figure 3.10-2 indicate a peak clad temperature of 2168°F corresponding to a 32°F margin to the 2200°F limit.[2]

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

In the specified limit of $F_{\Delta H}^N$ there is a 8 percent allowance for uncertainties which means that normal operation of the core is expected to result in $F_{\Delta H}^N \leq 1.55/1.08$. The logic behind the larger uncertainty in this case is that (a) normal perturbations in the radial power shape

(e.g. rod misalignment) affect $F^N_{\Delta H}$, in most cases without necessarily affecting F_Q, (b) the operator has a direct influence on F_Q through movement of rods, and can limit it to the desired value, he has no direct control over $F^N_{\Delta H}$ and (c) an error in the predictions for radial power shape, which may be detected during startup physics tests can be compensated for in F_Q by tighter axial control, but compensation for $F^N_{\Delta H}$ is less readily available. When a measurement of $F^N_{\Delta H}$ is taken, experimental error must be allowed for and 4 percent is the appropriate allowance for a full core map taken with the moveable incore detector flux mapping system.

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

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

1. Control rods in a single bank move together with no individual rod insertion differing by more than 15 inches from the bank demand position. An indicated misalignment limit of 13 steps precludes a rod misalignment no greater than 15 inches with consideration of maximum instrumentation error.

2. Control Rod banks are sequenced with overlapping banks as described in Technical Specification 3.10.4.

3. The full length and part length control bank insertion limits are not violated.

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

The permitted relaxation in $F^N_{\Delta H}$ allows radial power shape changes with rod insertion to the insertion limits. It has been determined that provided the above conditions 1 through 4 are observed, these hot channel factors limits are met. In Specification 3.10.2, F_Q is arbitrarily limited for $P \leq 0.5$ (except for low power physics tests).

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

The technical specifications on power distribution control assure that F_Q upper bound envelope of 2.32 times Figure 3.10-2 is not exceeded and xenon distributions are not developed which at a later time, would cause greater local power peaking even though the flux difference is then within the limits specified by the procedure.

The target (or reference) value of flux difference is determined as follows. At any time that equilibrium xenon conditions have been established, the indicated flux difference is noted with part length rods withdrawn from the core and with the full length rod control rod bank more than 190 steps withdrawn (i.e. normal full power operating position appropriate for the time in life, usually withdrawn farther as burnup

proceeds). This value, divided by the fraction of full power at which the core was operating is the full power value of the target flux difference. Values for all other core power levels are obtained by multiplying the full power value by the fractional power. Since the indicated equilibrium value was noted, no allowances for excore detector error are necessary and indicated deviation of $\pm$ 5% ΔI are permitted from the indicated reference value. During periods where extensive load following is required, it may be impractical to establish the required core conditions for measuring the target flux difference every month. For this reason, the specification provides two methods for updating the target flux difference.

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

In some instances of rapid plant power reduction, automatic rod motion will cause the flux difference to deviate from the target band when the reduced power level is reached. This does not necessarily affect the xenon distribution sufficiently to change the envelope of peaking factors which can be reached on a subsequent return to full power within the target band. However, to simplify the specification, a limitation of one hour in any period of 24 hours is placed on operation outside the band. This ensures that the resulting xenon distributions are not significantly different from those resulting from operation within the target band. The instantaneous consequences of being outside the band, provided rod insertion limits are observed, is not worse than a 10 percent increment in peaking factor for flux difference in the range +14 to -14 percent (+11 percent to

-11 percent indicated) increasing by ± 1 percent for each 2 percent decrease in rated power. Therefore, while the deviation exists the power level is limited to 90 percent or lower, depending on the indicated flux difference.

If, for any reason, flux difference is not controlled within the ± 5 percent band for as long a period as one hour, then xenon distributions may be significantly changed and operation at 50 percent is required to protect against potentially more severe consequences of some accidents.

As discussed above, the essence of the procedure is to maintain the xenon distribution in the core as close to the equilibrium full power condition as possible. This is accomplished without part length rods by using the boron system to position the full length control rods to produce the required indicated flux difference.

For FSAR Section 14.1 events, the core is protected from overpower and a minimum DNBR of 1.30 by an automatic protection system. Compliance with operating procedures is assumed as a precondition for FSAR Section 14.1 events. However, operator error and equipment malfunctions are separately assumed to lead to the cause of the transients considered.

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

described below.

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

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

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

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

If tilt ratio greater than 1.09 occurs which is not due to a misaligned rod, the reactor shall be brought to a hot shutdown condition for investigation. However, if the tilt condition can be identified as due to rod misalignment, operation can continue at a reduced power (2% for each one percent the tilt ratio exceeds 1.0) for two hours to correct the rod misalignment.

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

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

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

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

The rod position indicator channel is sufficiently accurate to detect a rod ±7 inches away from its demand position. An indicated misalignment less than 13 steps does not exceed the power peaking factor limits. If the rod position indicator channel is not operable, the operator will be fully aware of the inoperability of the channel, and special surveillance of core power tilt indications, using established procedures and relying on excore nuclear detectors, and/or movable incore detectors, will be used to verify power distribution symmetry. These indirect measurements do not have the same resolution if the bank is near either end of the core, because a 13 step misalignment would have no effect on power distribution. Therefore, it is necessary to apply the indirect checks following significant rod motion.

One inoperable control rod is acceptable provided that the power distribution limits are met, trip shutdown capability is available, and provided the potential hypothetical ejection of the inoperable rod is not worse than the cases analyzed in the safety analysis report. The rod ejection accident for an isolated fully inserted rod will be worse if the residence time of the rod is long enough to cause significant non-uniform fuel depletion. The 4 week period is short compared with the time interval required to achieve a significant non-uniform fuel depletion.

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

REFERENCE

1. WCAP-8576, "Augmented Startup and Cycle 1 Physics Program", August 1975
2. FSAR Appendix 14C

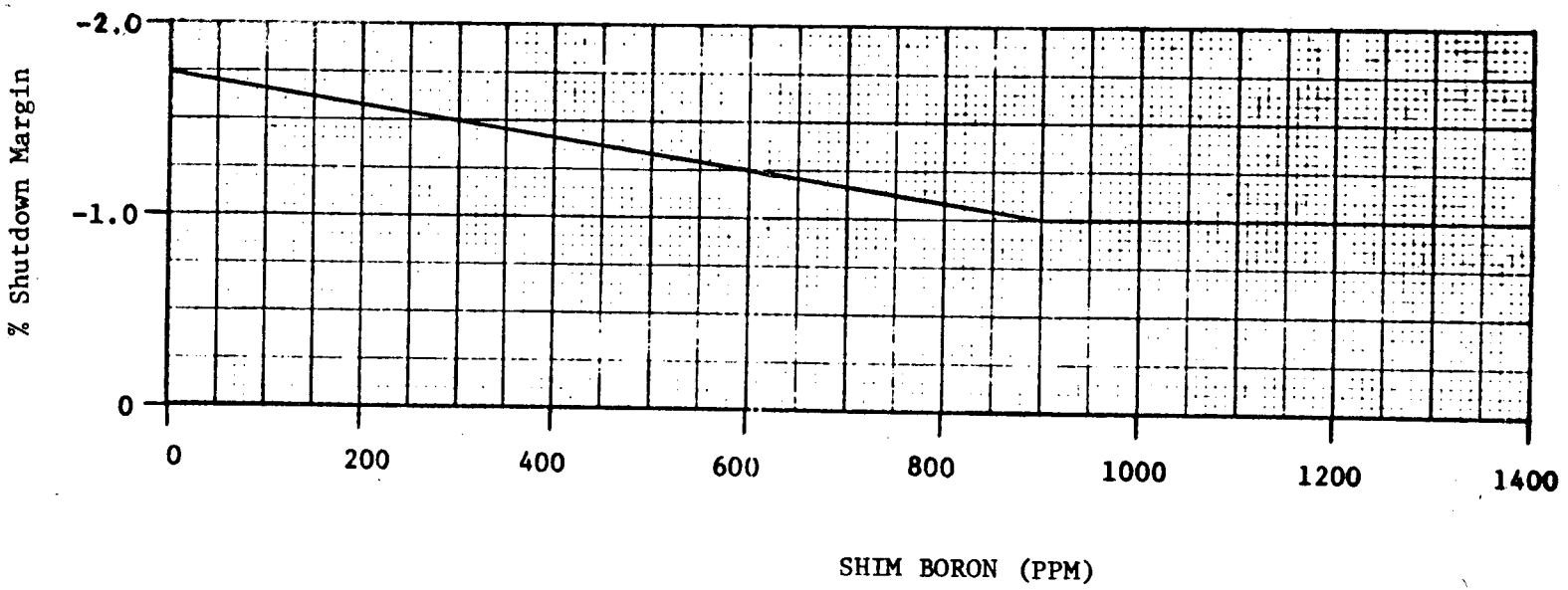


Figure 3.10-1 Required Shutdown Margin

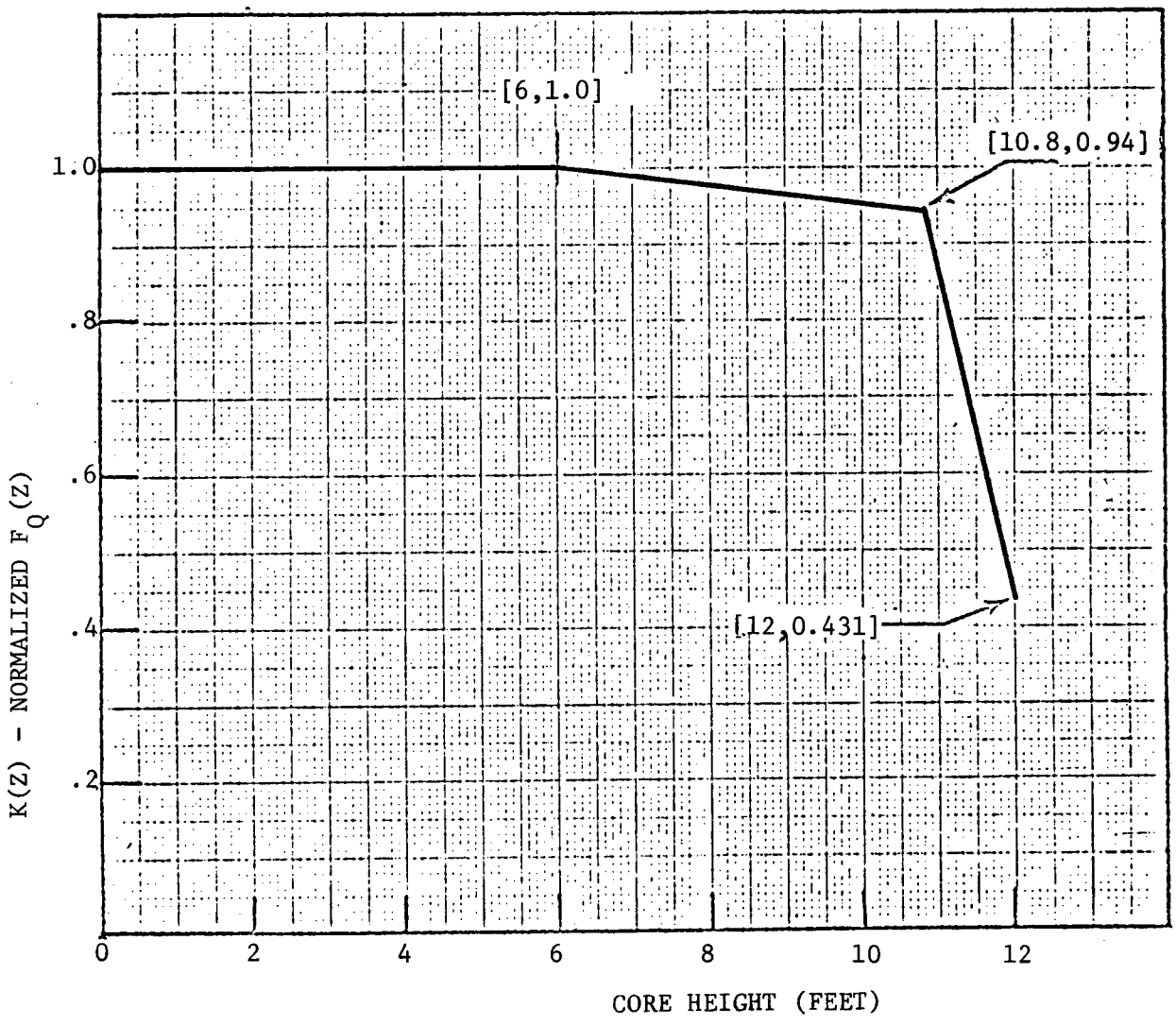


Figure 3.10-2 Hot Channel Factor Normalized Operating Envelope

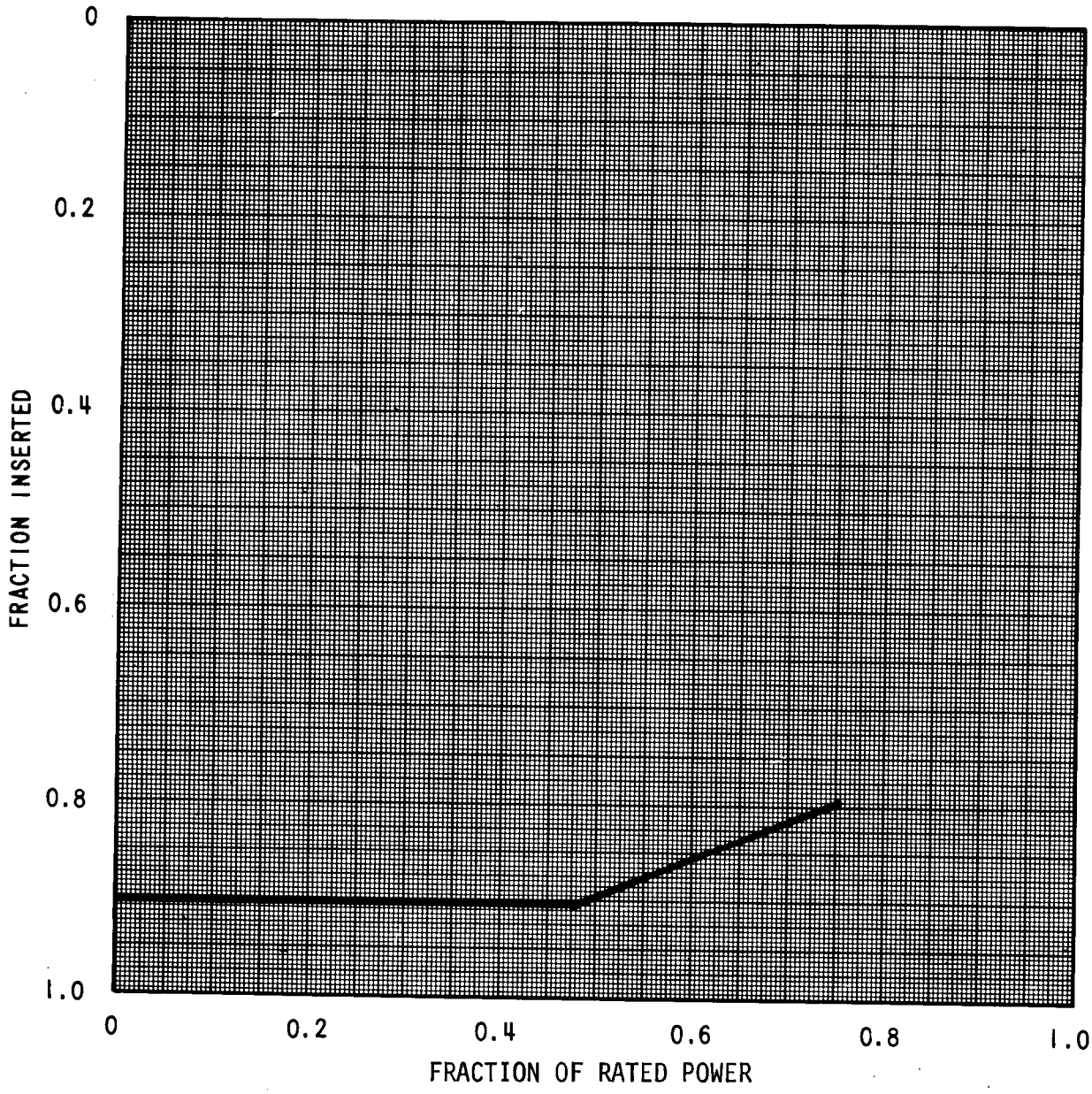
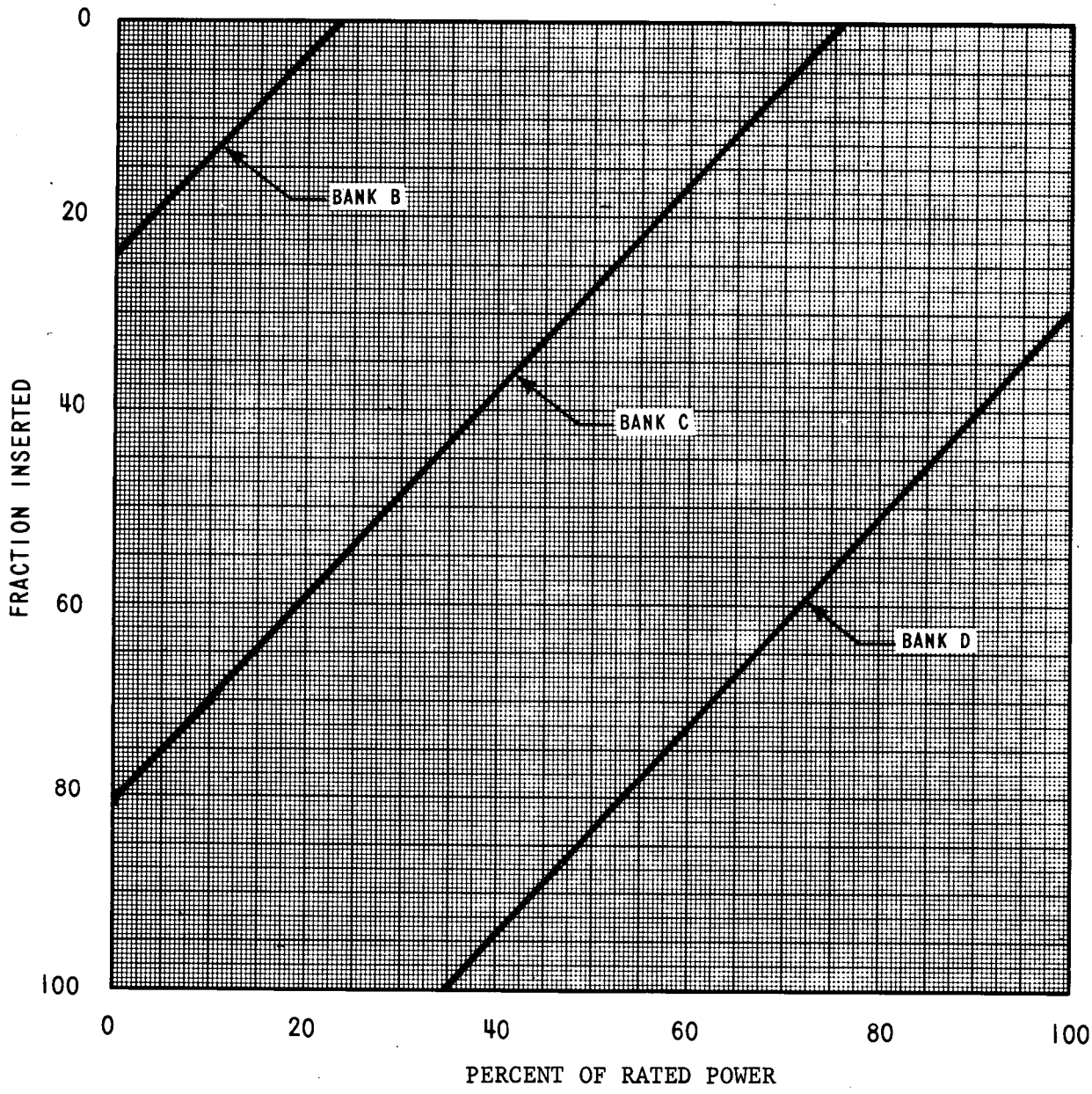
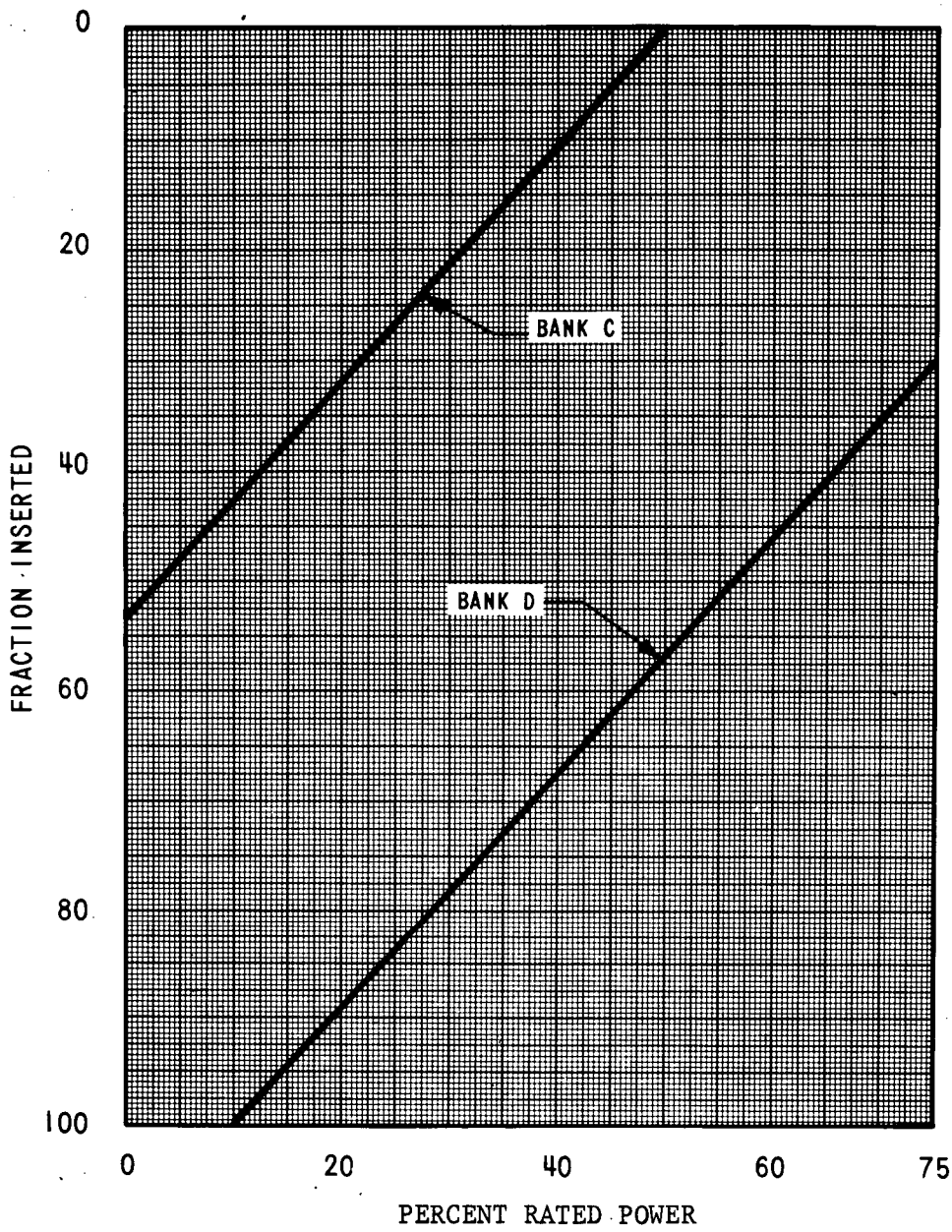


Figure 3.10-3. Part Length Rod Insertion Limit vs. Power



NOTE: BANK A IS FULLY WITHDRAWN AT ZERO POWER

Figure 3.10-4. Full Length Rod Insertion Limits 100 Step Overlap Four Loop Operation



NOTE: BANKS A & B ARE FULLY WITHDRAWN AT ZERO POWER

Figure 3.10-5. Full Length Rod Insertion Limits 100 Step Overlap 3 Loop Operation

3.11 MOVABLE IN-CORE INSTRUMENTATION

Applicability

Applies to the operability of the movable detector instrumentation system.

Objective

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

Specification

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

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

Basis

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

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

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

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

Reference

(1) FSAR - Section 7.4

3.12 RIVER LEVEL

Applicability

Applies to water elevation of the Hudson River as measured at the Indian Point Unit No. 3 intake structure.

Objective

To specify the maximum water elevation of the Hudson River for safe operation of the reactor.

Specification

When the Hudson River water elevation as measured at the Indian Point Unit No. 3 intake structure reaches 11'-0" above mean sea level, sandbagging the service water pumps will be initiated. If the Hudson River water elevation reaches 12'-5" above mean sea level at the Indian Point Unit No. 3 intake structure, the reactor will be brought to a cold shutdown condition using normal operating procedures.

Basis

Analyses have been performed which indicate that the river water elevation would have to reach 15'-3" above mean sea level before it would seep into the lowest floor elevation of any of the buildings housing equipment vital for safe shutdown of the reactor.[1] Monitoring of the Hudson River water elevation will not be required until there is a flood warning notice disseminated by the New York City National Oceanographic and Atmosphere Administration (NOAA) office.

References:

(1) FSAR, Section 2.5

4 SURVEILLANCE REQUIREMENTS

4.1 OPERATIONAL SAFETY REVIEW

Applicability

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

Objective

To specify the minimum frequency and type of surveillance to be applied to plant equipment and conditions. Performance of any surveillance test outlined in these specifications is not required if the plant condition is the same as the condition into which the plant would be placed by an unsatisfactory result of that test.

Specification

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

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

Basis

A surveillance test is intended to identify conditions in a plant that would lead to a degradation of reactor safety. Should a test reveal such a condition, then the Technical Specifications require that, either immediately or after a specified period of time, the plant be placed in a condition which mitigates or eliminates the consequences of additional related casualties or accidents. If the plant is already in a condition which would satisfy the failure criteria of the test, then plant safety is assured and performance of the test yields either meaningless information or information that is not necessary to determine safety limits or limiting conditions for operation of the plant.

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

Calibration

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

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

Other channels are subject only to the "drift" errors induced within the instrumentation itself and, consequently, can tolerate longer intervals between calibration. Process system instrumentation errors induced by drift can be expected to remain within acceptable tolerances if recalibration is performed at intervals of each refueling shutdown.

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

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

Testing

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

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

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

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

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

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

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

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

TABLE 4.1-1 (Sheet 1 of 3)

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTS OF INSTRUMENT CHANNELS

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

* By means of the movable incore detector system

** Monthly when reactor power is below the setpoint and prior to each startup if not done previous month.

TABLE 4.1-1 (Sheet 2 of 3)

Channel Description	Check	Calibrate	Test	Remarks
10. Steam Generator Level	S	R	M	
11. Residual Heat Removal Pump Flow	N.A.	R	N.A.	
12. Boric Acid Tank Level	S	R	N.A.	Bubbler tube rodded during calibration
13. Refueling Water Storage Tank Level	W	R	N.A.	Low level alarms
14. (a) Containment Pressure	S	R	M	High
(b) Containment Pressure	S	R	M	High High
15. Process and Area Radiation Monitoring Systems	D	R	Q	
16. Containment and Recirculation Sump Level	N.A.	N.A.	R	
17. Accumulator Level and Pressure	S ***	R	N.A.	
18. Steam Line Pressure	S	R	M	
19. Turbine First Stage Pressure	S	R	M	
20. Logic Channel testing	N.A.	N.A.	M	
21. Turbine Overspeed Protection Trip Channel (Electrical)	N.A.	R	M	
22. Boron Injection Tank Return Flow	S	R	N.A.	

*** If either an accumulator level or pressure instrument channel is declared inoperable, the remaining level or pressure channel must be verified operable by interconnecting and equalizing (Pressure and/or level wise) a minimum of two accumulators and cross-checking the instrumentation.

TABLE 4.1-1 (Sheet 3 of 3)

	Channel Description	Check	Calibrate	Test	Remarks
23.	Temperature Sensors in Auxiliary Boiler Feedwater Pump Building	N.A.	N.A.	R	
24.	Temperature Sensors in Penetration Area of Primary Auxiliary Building	N.A.	N.A.	R	
25.	Level Sensors in Turbine Building	N.A.	N.A.	R	
26.	Volume Control Tank Level	N.A.	R	N.A.	
27.	Boric Acid Make-Up Flow Channel	N.A.	R	N.A.	

S - Each Shift
D - Daily
W - Weekly
M - Monthly
P - Prior to each startup if not done previous week
Q - Quarterly
R - Each Refueling outage
NA - Not Applicable

TABLE 4.1-2 (Sheet 1 of 2)

FREQUENCIES FOR SAMPLING TESTS

	Sample	Analysis	Frequency	Maximum Time Between Analysis
1.	Reactor Coolant	Gross Activity(1)	5 days/week(1) (4)	3 days(4)
		Tritium Activity	Weekly(1)	10 days
		Boron concentration	2 days/week	5 days
		Radiochemical (gamma)(2) Spectral Check	Monthly	45 days
		Oxygen and Chlorides Concentration	3 times per 7 days	3 days
		Fluorides Concentration	Weekly	10 days
		$\bar{E}$ Determination(3)	Semi-Annually	30 weeks
		Isotopic Analysis for I-131, I-133, I-135	Once per 14 days(5)	20 days
2.	Boric Acid Tank/Boron Injection Tank	Boron Concentration, Chlorides	Weekly	10 days
3.	Spray Additive Tank	NaOH Concentration	Monthly	45 days
4.	Accumulators	Boron Concentration	Monthly	45 days
5.	Refueling Water Storage Tank	Boron Concentration pH, Chlorides	Monthly	45 days
		Gross Activity	Quarterly	16 weeks
6.	Secondary Coolant	I-131 Equivalent (Isotopic Analysis)	Monthly	45 days
		Gross Activity	3 times per 7 days	3 days
7.	Component Cooling Water	Gross Activity, Corrosion Inhibitor and pH	Monthly	45 days
8.	Spent Fuel Pool (when fuel stored)	Gross Activity Boron Concentration, Chlorides	Monthly	45 days

TABLE 4.1-2 (Sheet 2 of 2)

FREQUENCIES FOR SAMPLING TESTS

FOOTNOTES:

(1) A gross activity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units of μCi/cc.

(2) A radiochemical analysis shall consist of the quantitative measurement of each radionuclide with half life greater than 10 minutes making up at least 95% of the total activity of the primary coolant.

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

(4) Whenever the Gross Failed Fuel Monitor is inoperable, the sampling frequency shall be increased to twice per day, five days per week. The maximum time between analyses shall be sixteen hours for the two samples taken on a given day and three days between daily analysis. This accelerated sampling frequency need only be performed until the Gross Failed Fuel Monitor is declared operable.

(5) Once per 4 hours whenever the DOSE EQUIVALENT I-131 exceeds 1.0 μCi/cc or one sample after two hours but before six hours following a thermal power change exceeding 15 percent of the rated thermal power within a one-hour period.

TABLE 4.1-3 (Sheet 1 of 1)

FREQUENCIES FOR EQUIPMENT TESTS

		Check	Frequency
1.	Control Rods	Rod drop times of all full length rods	R
2.	Control Rod	Partial movement of all full length rods	Every 2 weeks during reactor critical operations
3.	Pressurizer Safety Valves	Set point	R
4.	Main Steam Safety Valves	Set point	R
5.	Containment Isolation System	Automatic actuation	R
6.	Refueling System Interlocks	Functioning	Prior to each refueling outage
7.	Fire Protection System and Power Supply	Functioning	Annually
8.	Primary System Leakage	Evaluate	5 days/week
9.	Diesel Fuel Supply	Fuel Inventory	Weekly
10.	Turbine Steam Stop, Control Valves	Closure	Monthly
11.	L.P. Steam Dump System (6 Lines)	Closure	Monthly
12.	Service Water System	Each pump starts and operates for 15 minutes (unless already operating)	Monthly
13.	City Water Connections to Charging Pumps and Boric Acid Piping	Temporary connections available and valves operable	R

R Each refueling outage

4.2 PRIMARY SYSTEM SURVEILLANCE

4.2.1 APPLICABILITY

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

4.2.2 OBJECTIVE

To assure the continued integrity of the primary system boundary, and steam generator shells.

4.2.3 SPECIFICATION

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

b. Post-operational non-destructive inspections listed in Table 4.2-1. shall be performed as specified. The results obtained from compliance with this specification shall be evaluated after five years and the conclusions of this evaluation shall be reviewed with the NRC.

c. The structural integrity of the primary system boundary and steam generator shells shall be maintained at the level required by the original acceptance standards throughout the life of the plant. Any evidence as a result of the inspections listed in Table 4.2-1, that defects have initiated or grown shall be investigated, including evaluation of comparable areas of the primary system or steam generator shells.

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

4.2.4

The inspection interval shall be ten years.

4.2.5

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

a. UT - Ultrasonic examination per paragraph IS-213.2.*
b. RT - Radiographic examination per paragraph IS-213.1.
c. MT - Magnetic particle examination per paragraph IS-212.1.
d. PT - Liquid penetrant examination per paragraph IS-212.2.
e. V - Visual examination per paragraphs IS-211.1 or IS211.2.

* All indications which produce a response greater than 100% of the reference level shall be investigated to the extent that the operator can determine the shape, identity, and location of all such reflectors and evaluate these indications per paragraph IS-311.

4.2.6

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

4.2.7

With the exception of those components or areas for which the examination may be deferred to the end of the inspection interval, at least 25 percent of the required examinations shall have been completed by

the expiration of one-third of the inspection interval (with credit for no more than 33-1/3 percent if additional examinations are completed) and at least 50 percent shall have been completed by the expiration of two-thirds of the inspection interval (with credit for no more than 66-2/3 percent). The remaining required examinations shall be completed by the end of the inspection interval. Successive inspections shall meet the requirements of Paragraph IS-243 of the ASME Rules for In-Service Inspection of Nuclear Reactor Coolant Systems.

4.2.8 BASES

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

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

The inspection requirements of this section shall apply to all pressure-containing components that are part of the system boundary defined herein. Due to the design of Indian Point Unit #3, there may be areas where weld access is impossible due to high radiation and/or physical access problems. Exception is taken to performing inspections in these areas.

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

a. part of the reactor coolant system[1] or

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

(1) the outermost containment isolation valve[2] in system piping which penetrates primary reactor containment.

(2) the second of two valves[3] normally closed during normal reactor operation in system piping which does not penetrate primary reactor containment and

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

The system boundary also includes the steam generator shells.

Exclusions

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

(2) Components that can be isolated from the reactor coolant system by two valves (both closed, both open or one closed and the other open). Each valve must be capable of automatic actuation and its closure time must be such that, for postualted failure of the component during normal reactor operation (and the other valve is open), the reactor can be shut down and cooled down in an orderly manner assuming makeup is provided by the reactor coolant makeup system[4] only.

4.2.9

The examinations scheduled are listed in Table 4.2-1 and are discussed below:

A. Reactor Vessel and Closure Head

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

Due to the Indian Point Unit No. 3 plant design, the welds in the reactor vessel are not accessible from the O.D. It is intended that these welds be volumetrically examined from the I.D. when required, using remote, mechanized techniques. Since the examination of these welds requires removal of the core internals and thermal shield, the examinations are scheduled near the end of the ten-year inspection interval.

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

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

It is intended that these welds be volumetrically examined from the I.D., using remote, mechanized techniques. The extent of these examinations is predicated on the capability of the remote mechanized equipment.

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

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

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

Due to the Indian Point Unit No. 3 plant design, the vessel to flange weld in the reactor vessel is not accessible from the O.D. It is intended that this weld be volumetrically examined from the I.D. using remote mechanized techniques. The examinations scheduled to be performed on these welds are shown in Table 4.2-1.

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

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

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

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

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

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

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

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

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

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

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

There are no pressure-retaining bolts less than two inches on the Indian Point Unit No.3 vessel.

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

There are a total of four vessel support pads welded to inlet and outlet nozzles on the Indian Point Unit No. 3 vessel. In accordance with Category H of Table IS-251 of the Code, the area to be examined should be the weld connection between the nozzle and the vessel shell. This examination is covered by Item 1.4 above.

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

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

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

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

B. Pressurizer

ITEM 2.1 (CATEGORY B) - Longitudinal and Circumferential Welds

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

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

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

ITEM 2.3 (CATEGORY E-1) - Heater Connections

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

ITEM 2.4 (CATEGORY E-2) - Heater Connections

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

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

There is no pressure-retaining bolting on the Indian Point Unit No. 3 pressurizer two inches or greater in diameter.

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

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

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

The integrally vessel support skirt weld will be inspected in accordance with the requirements of Section XI.

ITEM 2.8 (CATEGORY I-2) - Vessel Cladding

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

C. Steam Generator

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

The primary head for the Indian Point Unit No. 3 steam generators is a one-piece casting. Thus, the only weld covered by this category is the tubesheet-to-head weld. It is believed that this weld can be examined from the O.D. of the steam generator by ultrasonic techniques. The examinations scheduled for this weld are shown in Table 4.2-1.

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

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

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

The steam generator safe-ends are a buttered end preparation of the cast nozzle and are located between the nozzle and cast fittings. These safe end welds will be examined in accordance with Section XI of the ASME Code (January, 1970).

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

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

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

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

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

Indian Point Unit No. 3 generator supports are not integrally welded to the steam generator. Thus, this item does not apply.

ITEM 3.7 (CATEGORY I-2) - Vessel Cladding

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

ITEM 3.7 (CATEGORY I-2) - Steam Generator Head - Weld Cladding

The exterior surfaces of the head are accessible by removal of covering insulation. The surfaces to be examined by ultrasonic testing during surveillance test operation have been prepared and specific location of test sites are noted in Figures 4.2-1 and 4.2-2. The interior areas of the primary sides of the steam generator are accessible through manways provided on the inlet and outlet sides of the head.

Visual examination of the interior surfaces of the head will be performed and will be recorded on videotape at each of the first three refueling shutdowns. Evaluation of the results of the prior examinations will be utilized to establish the scope of subsequent visual examinations.

Specific sections of the internal weld surfaces of the head, which after visual examination require more definitive investigation, shall be evaluated by liquid penetrant testing. Additional non-destructive testing including replication using RTV-11 silicone rubber type material will be employed where liquid penetrant results indicate that they are required.

Monitoring of the existing welded cladding conditions shall be performed by ultrasonic testing from the exterior surface. Eight sectors on three steam generator heads at the locations shown in Figures 4.2-1 and 4.2-2 shall be surveyed at each of the first three refueling outages using the ultrasonic test process previously qualified. The results will be compared to base line data generated prior to reactor startup. Evaluation of the results following the third refueling outage will determine whether a test program should be continued for subsequent shutdowns.

ITEM 3.8 - Steam Generator Shell Welds

The areas to be examined are representative shell and head circumferential welds which are gross structural discontinuities[5]. The area shall include weld metal and base metal for one plate thickness beyond the edge of the weld joint.

Over the service life time, for each weld inspected examine 20 percent of the weld uniformly distributed, where accessible, among three areas around the vessel circumference.

D. Piping Pressure Boundary

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

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

ITEM 4.2 (CATEGORY J-1) - Circumferential and Longitudinal Pipe Welds and Branch Pipe Connections Welds Larger Than 4 inches in Diameter

Due to the design of the Indian Point Unit No. 3 piping systems, there may be areas where access to piping welds will be impossible due to high radiation levels and/or physical access problems. Existing examination techniques may also limit inspections. Exception is taken to performing inspections in these areas. The remaining welds in the primary system will be examined in such a manner as to cumulatively cover 25% of the welds during the inspection interval. The examinations scheduled are given in Table 4.2-1.

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

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

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

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

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

The accessible integrally-welded supports in the Indian Point Unit No. 3 piping systems subject to this inspection will be examined in accordance with the schedule shown in Table 4.2-1.

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

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

ITEM 4.7 (CATEGORY J-2) - Circumferential and Longitudinal Pipe Welds and Branch Pipe Connections Welds

Due to the design of the Indian Point Unit No. 3 piping systems, there may be areas where access to piping welds will be impossible due to high radiation levels and/or physical access problems. Exception is taken to performing inspections in these areas.

The accessible welds in the Indian Point Unit No. 3 piping systems subject to this inspection will be examined in accordance with Section XI of the Code. The examinations scheduled are shown in Table 4.2-1.

ITEM 4.8 (CATEGORY J-1) - Socket Welds and Pipe Branch Connections Welds 4 inches Diameter and Smaller

Due to the design of the Indian Point Unit No. 3 piping systems, there may be areas where access to piping welds will be impossible due to high radiation levels and/or physical access problems. Exception is taken to performing inspections on socket welds within the primary boundary and sampling and instrumentation piping and thermowells.

The remaining welds in the primary system will be examined in such a manner as to cumulatively cover 25% of the welds during the inspection interval. The examinations scheduled are given in Table 4.2-1.

E. Pump Pressure Boundary

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

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

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

ITEM 5.2 (CATEGORY L-2) - Pump Casing

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

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

There are no nozzle-to-safe-end welds on the Indian Point Unit No. 3 reactor coolant pumps.

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

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

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

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

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

The reactor coolant pump supports consist of a cast foot welded to the pump casing. There are no currently known techniques for volumetrically inspecting these welds because of their geometric configuration.

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

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

F. Valve Pressure Boundary

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

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

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

ITEM 6.2 (CATEGORY M-2) - Valve Bodies

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

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

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

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

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

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

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

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

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

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

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

G. Miscellaneous Inspections

ITEM 7.1 - Primary Pump Flywheels

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

ITEM 7.2 - Materials Irradiation Surveillance Specimens

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

Capsule T	Replacement of first region of core
Capsule Y	5 years
Capsule Z	10 years
Capsule S	20 years
Capsule V	30 years
Capsule U	Standby
Capsule W	Standby
Capsule X	Standby

Reference:

(*) FSAR Section 4.5

NOTES

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

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

(3) Two check valves in series are acceptable.

(4) See Note 1 on Page 4 of Section XI.

(5) Gross structural discontinuity is a geometric or material discontinuity which affects the strain or stress distribution through the entire wall thickness of the component as defined in subparagraph NB-3213.2 of ASME Boiler and Pressure Vessel Code, Section III Nuclear Power Plant Components.

TABLE 4.2-1 (Sheet 1 of 12)

INSERVICE INSPECTION REQUIREMENTS FOR INDIAN POINT UNIT NO. 3

Item No.	Examination Category*	Components and Parts to be Examined	Method	Extent of Examination (Percent in 10 Year Interval)	Remarks
		REACTOR VESSEL AND CLOSURE HEAD			
1.1	A	Longitudinal and circumferential shell welds in core region	UT	Longitudinal - 10% Circumferential - 5%	
1.2	B	Longitudinal and circumferential welds in shell (other than those of Category A and C), and meridional and circumferential seam welds in bottom head and closure head (other than those of Category C)	UT	Longitudinal - 10% Circumferential - 5%	These inspections will be conducted within the capability of the remote, mechanical ultrasonic examination devices.
1.3	C	Vessel-to-flange and head-to-flange circumferential welds	UT	100%	

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

TABLE 4.2-1 (Sheet 2 of 12)

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

TABLE 4.2-1 (Sheet 3 of 12)

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

TABLE 4.2-1 (Sheet 4 of 12)

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

TABLE 4.2-1 (Sheet 5 of 12)

Item No.	Examination Category	Components and Parts to be Examined	Method	Extent of Examination (Percent in 10 Year Interval)	Remarks
2.4	E-2	Heater connections and instrument and sample nozzles.	V	See remarks	Visual inspections for leakage will be performed on at least 25% of the penetrations during the system hydrostatic test.
2.5	G-1	Pressure-retaining bolting		Not applicable.	
2.6	G-2	Pressure-retaining bolting.	V	100%	
2.7	H	Integrally-welded vessel supports	V & UT	10% of linear feet of weld	The inspection is limited to circumferential weld attaching the support skirt to the vessel.
2.8	I-2	Vessel Cladding	V	1 Patch	One (1) patch (36 square inches) on the primary side near the manway will be examined during the ten-year interval.
		HEAT EXCHANGERS (CLASS A) AND STEAM GENERATORS			
3.1	B	Longitudinal and circumferential welds, including tube-sheet-to-head or shell welds on the primary side.	V & UT	5% See remarks	The inspection is limited to the circumferential weld attaching the tube sheet to the lower head.

TABLE 4.2-1 (Sheet 6 of 12)

Item No.	Examination Category	Components and Parts to be Examined	Method	Extent of Examination (Percent in 10 Year Interval)	Remarks
3.2	D	Primary nozzle-to-vessel head welds and nozzle-to-head inside radiused section		See remarks	The primary nozzles are cast with the head. No inspections are planned.
3.3	F	Primary nozzle-to-safe-end welds	UT, V & PT	100%	
3.4	G-1	Pressure-retaining bolting		Not applicable	
3.5	G-2	Pressure-retaining bolting	V	100%	
3.6	H	Integrally-welded vessel supports		Not applicable	
3.7	I-2	Vessel cladding	V	1 patch	One (1) patch (36 square inches) in each primary side will be examined during the ten-year interval.

TABLE 4.2-1 (Sheet 7 of 12)

Item No.	Examination Category	Components and Parts to be Examined	Method	Extent of Examination	Remarks
3.7	I-2	Steam Generator Head welded cladding surfaces - primary sides.	V	100%	To be performed at each of the first three refueling shutdowns. The 100% visual inspection shall be recorded on videotape.
		"	PT	Sample	As warranted by results of visual examination.
		"	Alternate NDT or replication by RTV-11		As warranted by results of visual and liquid penetrant examinations.
		Steam Generator Head welded cladding surfaces - primary sides and eight sectors on Steam Generators #31, #32 and #34 per Figures 4.2-1 and 4.2-2	UT		To be performed at each of the first three refueling shutdowns.

TABLE 4.2-1 (Sheet 8 of 12)

Item No.	Examination Category	Components and Parts to be Examined	Method	Extent of Examination (Percent in 10 Year Interval)	Remarks
3.8	-	Secondary side shell welds	UT	See Remarks	The total examination completed over the service life time will be equivalent of having performed 100% of the required examination.
		PIPING PRESSURE BOUNDARY			
4.1	F	Vessel, pump and valve safe-ends to primary pipe welds and safe-ends in branch piping welds.	UT, PT & V	100%	This examination covers only the pressurizer safe-ends.
4.2	J-1	Circumferential and longitudinal pipe welds and branch pipe connections welds larger than 4 inches in diameter	V & UT	25%	Exception is taken to inaccessible welds and welds where examination techniques limit inspections.
4.3	G-1	Pressure-retaining bolting		Not applicable	
4.4	G-2	Pressure-retaining bolting	V	100%	

TABLE 4.2-1 (Sheet 9 of 12)

Item No.	Examination Category	Components and Parts to be Examined	Method	Extent of Examination (Percent in 10 Year Interval)	Remarks
4.5	K-1	Integrally-welded supports	V & UT or PT	25%	Exception is taken for supports which are not accessible.
4.6	K-2	Piping support and hanger	V	100%	Exception is taken for those supports which are not accessible.
4.7	J-2	Circumferential and longitudinal pipe welds and branch pipe connections welds	V	100%	Exception is taken to inaccessible welds.
4.8	J-1	Socket welds and pipe branch connections welds 4 in. dia. and smaller	V & PT	25%	(1) Exception is taken to inaccessible welds. (2) Exception is taken for socket welds. (3) Exception is taken for sampling and instrumental piping and thermowells.

TABLE 4.2-1 (Sheet 10 of 12)

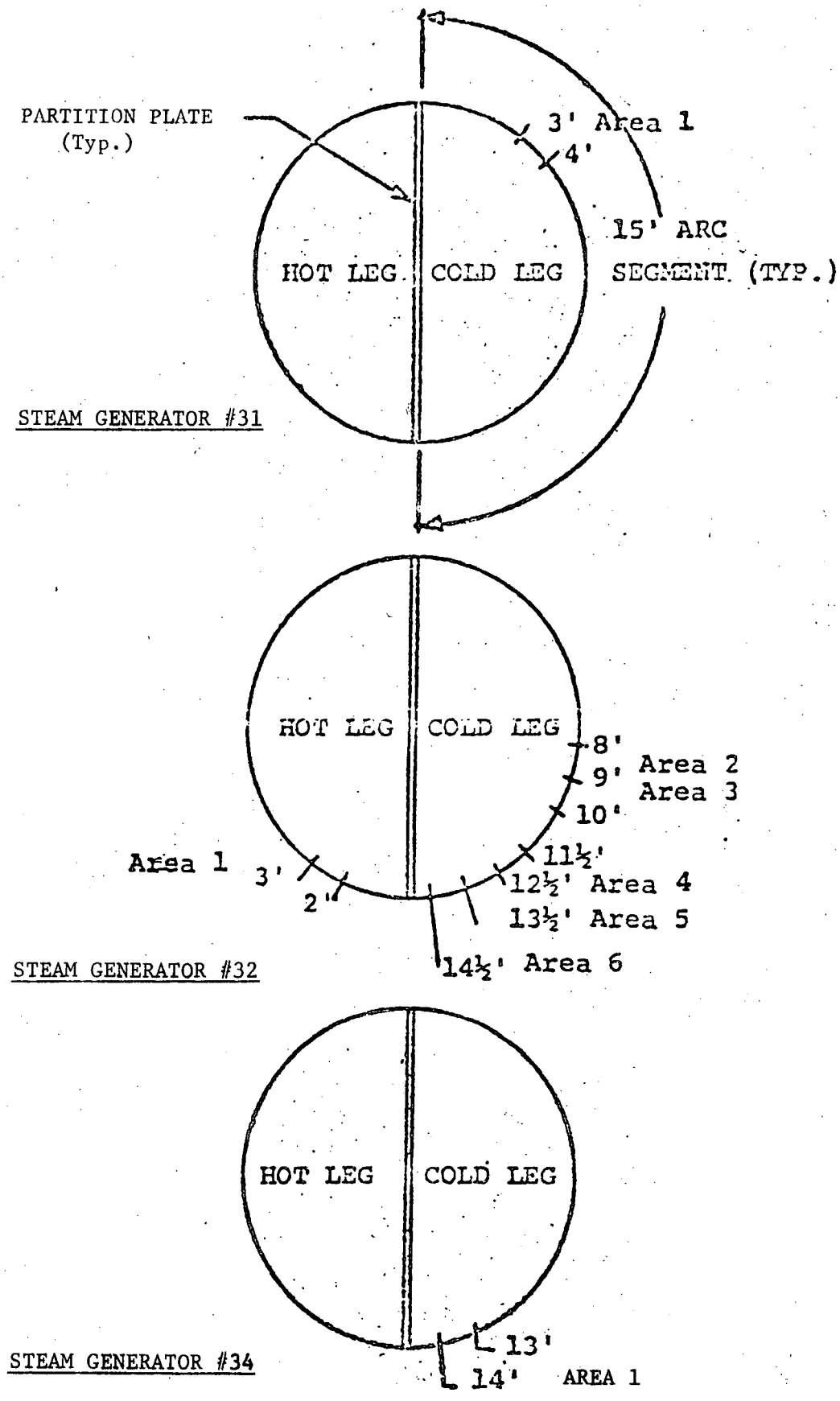
Item No.	Examination Category	Components and Parts to be Examined	Method	Extent of Examination (Percent in 10 Year Interval)	Remarks
		PUMP PRESSURE BOUNDARY			
5.1	L-1	Pump casing welds	V & PT	100%	No meaningful volumetric examinations can be performed on these welds.
5.2	L-2	Pump casings	V	See remarks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

TABLE 4.2-1 (Sheet 11 of 12)

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

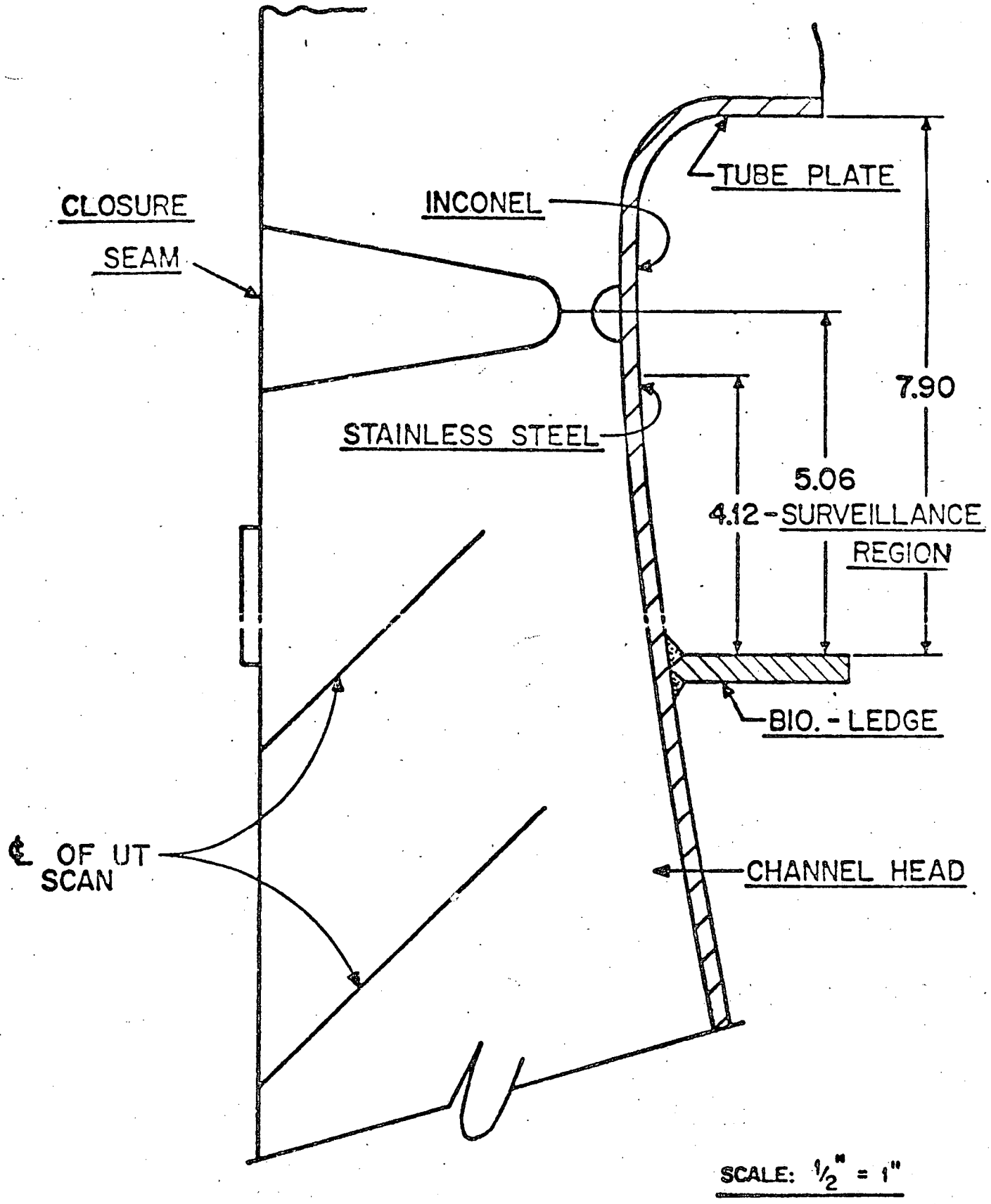
TABLE 4.2-1 (Sheet 12 of 12)

Item No.	Examination Category	Components and Parts to be Examined	Method	Extent of Examination (Percent in 10 Year Interval)	Remarks
6.4	G-1	Pressure-retaining bolting		Not applicable	
6.5	G-2	Pressure-retaining	V	100%	Exception is taken for valves which are not accessible.
6.6	K-1	Integrally-welded supports		Not applicable	
6.7	K-2	Supports and hangers	V	100%	Exception is taken for supports and hangers which are not accessible.
7.1		Primary pump flywheel	V & UT	See remarks	The flywheels shall be visually examined at the first refueling. At each subsequent refueling, one different flywheel shall be examined by ultrasonic methods.
7.2		Irradiation Specimen Schedule	Tensile and Charpy V Notch (Wedge Open Loading)	See remarks	Refer to FSAR Section 4.5.1.



STEAM GENERATOR PRIMARY SIDE ULTRASONIC TEST SECTIONS

FIGURE 4.2-1



SURVEILLANCE REGION

FIGURE 4.2-2

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 relationship shall be in accordance with the limits of Figure 4.3-1 for heatup for the first two-year period of operation (2 EFPY). 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: ± 100 psi is normal system pressure fluctuation), it will be leak tight during normal operation.

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

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 two 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 two 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. Figures 4.3-1 and 3.1-2 are recalculated periodically, using methods discussed in the Basis for Specification 3.1.B and results of surveillance specimens, as covered in Specification 4.2.

Reference

1. FSAR, Section 4.

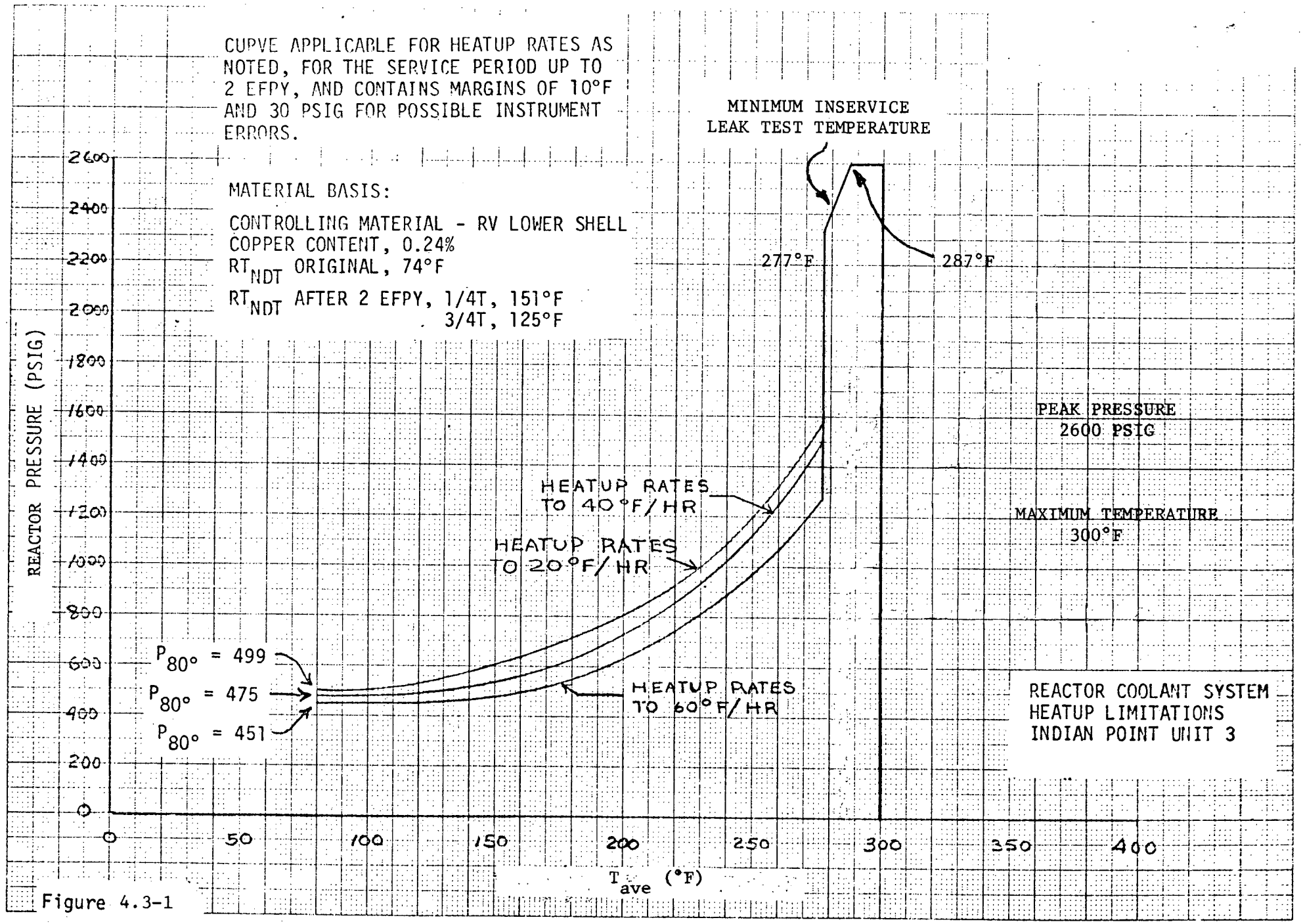


Figure 4.3-1

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. Integrated Leakage Rate

1. Test

a. A full pressure integrated leakage rate test shall be performed at intervals specified in A.3 at the peak accident pressure (P_a) of 40.6 psig minimum.

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

c. A general inspection of the accessible interior and exterior surfaces of the containment structures and components shall be performed prior to performing an integrated leak test to uncover any evidence of structural deterioration which may affect either the containment structural integrity or leak tightness. If there is evidence of structural deterioration, integrated leakage rate tests shall not be performed until corrective action is taken. Such structural deterioration and corrective actions taken shall be reported as part of the test report.

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

2. Acceptance Criteria

The measured leakage rate shall be less than 0.75 L_a where L_a is equal to 0.1 w/o per day of containment steam air atmosphere at 40.6 psig and 263°F, which are the peak accident pressure and temperature conditions.

3. Frequency

A set of three leakage rate tests shall be performed (during plant shutdown), at approximately equal intervals during each 10-year service period. The third test of each set shall be conducted when the plant is shutdown for the 10-year plant in service inspection.

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

1. Acceptance Criteria

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.

2. Corrective Action

a. If any time it is determined that the limit of B.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 shut down until repairs are effected and the continuous leakage meets the acceptance criterion.

C. Sensitive Leakage Rate

1. Test

A sensitive leakage rate test shall be conducted with the containment penetrations, weld channels, and certain double gasketed seals and isolation valve interspaces at a minimum pressure of 41 psig and with the containment building at atmospheric pressure.

2. Acceptance Criteria

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

3. Frequency

A sensitive leakage rate test shall be performed at a frequency of at least every other refueling but in no case at intervals greater than 3 years.

D. Air Lock Tests

1. The containment air locks shall be tested at a minimum pressure of 40.6 psig and at a frequency of every 6-months. The acceptance criteria is included in E.2.a.

2. Whenever containment integrity is required, verification shall be made of proper repressurization to at least 41 psig of the double-gasket air lock door seal upon closing an air lock door.

E. Containment Isolation Valves

1. Tests and Frequency

a. Isolation valves in Table 4.4-1 shall be tested for operability at a frequency of at least every refueling.

b. Isolation valves in Table 4.4-1 which are pressurized by the Weld Channel and Penetration Pressurization System shall be leakage tested as part of the Weld Channel and Penetration Pressurization System Test at a frequency of at least every refueling.

c. Isolation valves in Table 4.4-1 which are pressurized by the Isolation Valve Seal Water System shall be tested at a frequency of at least every refueling as part of an overall Isolation Valve Seal Water System Test.

d. Isolation valves in Table 4.4-1 which are not pressurized will be tested at a frequency of at least every refueling.

e. Isolation valves in Table 4.4-1 shall be tested with the medium and at the pressure specified therein.

2. Acceptance Criteria

a. The combined leakage rate for the following shall be less than 0.6 L_a: isolation valves listed in Table 4.4-1 subject to gas pressurization testing, air lock testing as specified in D.1, portions of the sensitive leakage rate test described in C.1 which pertain to containment penetrations and double-gasketed seals.

b. The leakage rate into containment for the isolation valves sealed with the service water system is 0.36 gpm per fan cooler.

F. 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 A.2, C.2, or E.2. Modifications or replacements performed directly prior to the conduct of an integrated leakage rate test shall not require a separate test.

G. Report of Test Results

Each integrated leakage rate test shall be the subject of a summary technical report to be submitted to the Nuclear Regulatory Commission in accordance with the requirements of Appendix J to 10 CFR 50, effective issue date March 16, 1973. Each report shall include leakage test results and a summary analyses of sensitive leak rate, air lock, and containment isolation valve tests performed since the previous integrated leakage rate test.

H. Annual Inspection

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

I. Residual Heat Removal System

1. Test

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

(2) The piping between the residual heat removal pumps suctions and the containment isolation valves in the residual heat removal pump suction line from the containment sump shall be hydrostatically tested at no less than 100 psig at the interval specified below.

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

2. Acceptance Criterion

The maximum allowable leakage from the Residual Heat Removal System components located outside of the containment shall not exceed two gallons per hour.

3. Corrective Action

Repairs or isolation shall be made as required to maintain leakage within the acceptance criterion.

4. Test Frequency

Tests of the Residual Heat Removal System shall be conducted at every refueling.

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

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

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

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

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

(b) conformance of the complete containment to a low leakage rate limit at 40.6 psig or higher during pre-operational testing, and

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

Secondly, the Weld Channel and Penetration Pressurization 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. A leak would be expected to build up slowly and would, therefore, be noted before design limits are exceeded. Remedial action can be taken before the limit is reached.

During normal plant operation, containment personnel lock door seals are continuously pressurized after each closure by the Weld Channel and Penetration Pressurization System. Whenever containment integrity is required, verification is made that seals repressurize properly upon closure of an air lock door.

A full pressure test of the air lock will be periodically performed at 6-month intervals to detect any unanticipated leakage.

The containment isolation valve leakage and sensitive leakage rate measurements obtained periodically, periodic inspection of accessible portions of the containment wall to detect possible damage to the liner plates, combined with the leakage monitoring afforded by the weld Channel and Penetration Pressurization System,[5] and IVSWS[6] provide assurance that the containment leakage is within design limits.

The testing of containment isolation valves in Table 4.4-1 either individually or in groups, utilizes the WC & PPS[5] or IVSWS[6] (where appropriate), and is in accordance with the requirements of Type C tests in Appendix J (issue effective date March 16, 1973) to 10CFR50. The specified test pressures are greater than the peak calculated accident pressure. Sufficient water is available in the Isolation Valve Seal Water System, Primary Water System, Service Water System, Residual Heat Removal System, and the City Water System to assure a sealing function for at least 30 days.

The acceptance criterion of 0.6 L_a for the combined leakage of isolation valves subject to gas pressurization, the air lock, containment penetrations and double-gasketed seals is in accordance with Appendix J (issue effective date March 16, 1973) to 10CFR50.

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

These specifications have been developed using Appendix J (issue effective date March 16, 1973) of 10CFR50 and ANSI N45.4-1972 "Leakage Rate Testing of Containment Structures for Nuclear Reactors" (March 16, 1972) for guidance.

The maximum permissible inleakage rate from the containment isolation valves sealed with service water for the full 12-month period of post accident recirculation without flooding the internal recirculation pumps is 0.36 gpm per fan cooler.

REFERENCES

(1) FSAR - Section 5
(2) FSAR - Section 5.1.7
(3) FSAR - Section 14.3.5
(4) FSAR - Volume 7, Response to Question 14.6
(5) FSAR - Section 6.6
(6) FSAR - Section 6.5

TABLE 4.4-1 (Page 1 of 7)

CONTAINMENT ISOLATION VALVES

Valve No.	Penetration Number(1)	Test Fluid(2)	Minimum Test Pressure (PSIG)
549	1	Water(4)	45
548	1	Water(4)	45
518	2	Gas	41
550	2	Gas	41
552	3	Water(4)	45
519	3	Water(4)	45
741	4	Water(5)	45(3)
744	4	Nitrogen(4)	41(3)
888A	5	Nitrogen(4)	41
888B	5	Nitrogen(4)	41
958	5	Nitrogen(4)	41
959	5	Nitrogen(4)	41
990C	5	Nitrogen(4)	41
1870	5	Nitrogen(4)	41
743	5	Nitrogen(4)	41
732	6	Nitrogen(4)	41(3)
885A	7	Water(5)	45
885B	7	Water(5)	45
201	8	Water(4)	45
202	8	Water(4)	45
205	9	Water(4)	45
226	9	Water(4)	45
227	9	Water(4)	45
250A	10	Water(4)	45
241A	10	Water(4)	45
250B	10	Water(4)	45
241B	10	Water(4)	45
250C	10	Water(4)	45

TABLE 4.4-1 (Page 2 of 7)

CONTAINMENT ISOLATION VALVES

Valve No.	Penetration Number (1)	Test Fluid (2)	Minimum Test Pressure (PSIG)
241C	10	Water (4)	45
250D	10	Water (4)	45
241D	10	Water (4)	45
222	11	Water (4)	45
956E	12	Water (4)	45
956F	12	Water (4)	45
869A	14	Water (4)	45
867A	14	Gas	41
878A	14	Gas	41
869B	14	Water (4)	45
867B	14	Gas	41
878B	14	Gas	41
1835A	15	Nitrogen (4)	41
1835B	15	Nitrogen (4)	41
1833A	15	Water (4)	45
1833B	15	Water (4)	45
851A	15	Water (4)	45
850A	15	Water (4)	45
859A	16	Water (4)	45
859C	16	Water (4)	45
891A	17	Gas	41
891B	17	Gas	41
891C	17	Gas	41
891D	17	Gas	41
863	17	Gas	41
956G	18	Water (4)	45
956H	18	Water (4)	45
1786	19	Water (4)	45
1787	19	Water (4)	45

TABLE 4.4-1 (Page 3 of 7)

CONTAINMENT ISOLATION VALVES

Valve No.	Penetration Number(1)	Test Fluid(2)	Minimum Test Pressure (PSIG)
1610	19	Gas	41
1616	19	Gas	41
1788	20	Water(4)	45
1789	20	Water(4)	45
1702	21	Water(4)	45
1705	21	Water(4)	45
797	22	Water(4)	45
769	22	Water(4)	45
784	23	Water(4)	45
786	23	Water(4)	45
FCV-625	24	Water(4)	45
789	24	Water(4)	45
791	29	Water(4)	45
798	29	Water(4)	45
796	30	Water(4)	45
793	30	Water(4)	45
1728	31	Water(4)	45
1723	31	Water(4)	45
1234	32	Gas(7)	41
1235	32	Gas(7)	41
1236	33	Gas(7)	41
1237	33	Gas(7)	41
PCV-1229	34	Gas(7)	41
PCV-1230	34	Gas(7)	41
PCV-1215	37	Water(4)	45
PCV-1215A	37	Water(4)	45
PCV-1214	37	Water(4)	45
PCV-1214A	37	Water(4)	45
PCV-1216	37	Water(4)	45
PCV-1216A	37	Water(4)	45
PCV-1217	37	Water(4)	45
PCV-1217A	37	Water(4)	45

TABLE 4.4-1 (Page 4 of 7)

CONTAINMENT ISOLATION VALVES

Valve No.	Penetration Number(1)	Test Fluid (2)	Minimum Test Pressure (PSIG)
PCV-1224	38	Water (4)	45
PCV-1224A	38	Water (4)	45
PCV-1223	38	Water (4)	45
PCV-1223A	38	Water (4)	45
PCV-1225	38	Water (4)	45
PCV-1225A	38	Water (4)	45
PCV-1226	38	Water (4)	45
PCV-1226A	38	Water (4)	45
SWN-41	39	Water (6)	45
SWN-43	39	Water (6)	45
SWN-42	39	Water (6)	45
SWN-41	39	Water (6)	45
SWN-43	39	Water (6)	45
SWN-42	39	Water (6)	45
SWN-41	39	Water (6)	45
SWN-43	39	Water (6)	45
SWN-42	39	Water (6)	45
SWN-41	39	Water (6)	45
SWN-43	39	Water (6)	45
SWN-42	39	Water (6)	45
SWN-41	39	Water (6)	45
SWN-43	39	Water (6)	45
SWN-42	39	Water (6)	45
SWN-44	40	Water (6)	45
SWN-51	40	Water (6)	45
SWN-44	40	Water (6)	45
SWN-51	40	Water (6)	45
SWN-44	40	Water (6)	45
SWN-51	40	Water (6)	45
SWN-44	40	Water (6)	45
SWN-51	40	Water (6)	45

TABLE 4.4-1 (Page 5 of 7)

CONTAINMENT ISOLATION VALVES

Valve No.	Penetration Number[1]	Test Fluid[2]	Minimum Test Pressure (PSIG)
SWN-44	40	Water[6]	45
SWN-51	40	Water[6]	45
SWN-71	40	Water[6]	45
SWN-71	40	Water[6]	45
SWN-71	40	Water[6]	45
SWN-71	40	Water[6]	45
SWN-71	40	Water[6]	45
SA-24	41	Water[4]	45
SA-24	41	Water[4]	45
580A	44	Gas	41
580B	44	Gas	41
UH-37	45	Water[4]	45
UH-38	46	Water[4]	45
1170	48	Gas[7]	41
1171	48	Gas[7]	41
1172	49	Gas[7]	41
1173	49	Gas[7]	41
1190	50	Gas[7]	41
1191	50	Gas[7]	41
1192	50	Gas[7]	41
990A	51	Nitrogen[4]	41
990B	51	Nitrogen[4]	41
956A	52	Water[4]	45
956B	52	Water[4]	45
956C	53	Water[4]	45
956D	53	Water[4]	45
1814A	54	Gas	41
1814B	55	Gas	41
1814C	56	Gas	41
1890D	57	Gas	41
1890E	57	Gas	41

TABLE 4.4-1 (Page 6 of 7)

CONTAINMENT ISOLATION VALVES

Valve No.	Penetration Number[1]	Test Fluid[2]	Minimum Test Pressure (PSIG)
1890A	57	Gas	41
1890C	57	Gas	41
1890F	57	Gas	41
1890B	57	Gas	41
1890G	57	Gas	41
1890H	57	Gas	41
1890J	57	Gas	41
1882A	58	Gas	41
IV-2A	58	Gas	41
IV-2B	58	Gas	41
1875A	59	Gas	41
IV-3A	59	Gas	41
1876A	60	Gas	41
IV-5A	60	Gas	41
1875B	61	Gas	41
IV-3B	61	Gas	41
1876B	62	Gas	41
IV-5B	62	Gas	41
IA-39	64	Gas	41
PCV-1228	64	Gas	41
PS-7	65	Gas[7]	41
PS-10	65	Gas[7]	41
PS-8	65	Gas[7]	41
PS-9	65	Gas[7]	41
CB-1	69	Gas	41
CB-2	69	Gas	41
CB-3	69	Gas[7]	41
CB-4	69	Gas[7]	41

TABLE 4.4-1 (Page 7 of 7)

CONTAINMENT ISOLATION VALVES

NOTES:

1. Reference: FSAR Table 5.2-1, Penetration No.

2. Gas Test Fluid indicates either nitrogen or air as test medium.

3. Testable only when at cold shutdown.

4. Isolation Valve Seal Water System.

5. Sealed by Residual Heat Removal System fluid.

6. Sealed by Service Water System.

7. Sealed by Weld Channel and Penetration Pressurization System.

4.5 TESTS FOR ENGINEERED SAFETY FEATURES AND AIR FILTRATION SYSTEMS

Applicability

Applies to testing of the Safety Injection System, the Containment Spray System, the Hydrogen Recombiner System, and the Air Filtration Systems.

Objective

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

Specification

A. SYSTEM TESTS

1. Safety Injection System

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

b. The test will be considered satisfactory if control board indication and visual observations indicate that all components have received the safety injection signal in the proper sequence and timing, that is, the appropriate pump breakers shall have opened and closed, and the appropriate valves shall have completed their travel.

c. Conduct a flow test of the high head safety injection system after any modification is made to either its piping and/or valve arrangement.

d. Verify that the mechanical stops on Valves 856 A, C, D, E, F, H, J and K are set at the position measured and recorded during the most recent ECCS operational flow test or flow tests performed in accordance with (c) above. This surveillance procedure shall be performed following any maintenance on these valves or their associated motor operators and at a convenient outage if the position of the mechanical stops have not been verified in the preceding three months.

2. Containment Spray System

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

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

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

3. Hydrogen Recombiner System

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

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

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

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

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

4. Containment Air Filtration System

a. Visual inspection of the filter installations shall be performed in accordance with ANSI N 510 (1975) every six months for the first two years and every refueling thereafter, or at any time fire, chemical releases or work done on the filters could alter their integrity.

b. At each refueling outage, the following conditions shall be demonstrated before the system can be considered operable:

(1) The pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches of water at ambient conditions and accident design flow rates.

(2) Using either direct or indirect measurements, the flow rate of the system fans shall be shown to be at least 90% of the accident design flow rate.

(3) The charcoal filter isolation valves shall be tested to verify operability.

c. At each refueling outage or at any time fire, chemical releases or work done on the filters could alter their integrity or after every 720 hours of charcoal adsorber use since the last test, the following conditions shall be demonstrated before the system can be considered operable:

(1) Impregnated activated charcoal from each of the five units shall have a methyl iodine removal efficiency $\geq$ 85% at $\pm$ 20% of the accident design flow rate, 5 to 15 mg/m^3 inlet methyl iodine concentration, $\geq$ 95% relative humidity and $\geq$ 250°F. In addition, ignition shall not occur below 300°F.

(2) A halogenated hydrocarbon (freon) test on charcoal adsorbers at ± 20% of the accident design flow rate and ambient conditions shall show ≥ 99% halogenated hydrocarbon removal.

(3) A locally generated DOP* test of the HEPA filters at ± 20% of the accident design flow rate and ambient conditions shall show ≥ 99% DOP removal.

5. Control Room Air Filtration System

a. Visual inspection of the filter installations shall be performed in accordance with ANSI N 510 (1975) every six months for the first two years and every refueling thereafter, or at any time fire, chemical releases or work done on the filters could alter their integrity.

b. The charcoal filtration system shall be operated for a minimum of 15 minutes every month.

c. At each refueling outage, the following conditions shall be demonstrated before the system can be considered operable:

(1) The pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches of water at ambient conditions and accident design flow rates.

(2) Using either direct or indirect measurements, the flow rate of the system fans shall be shown to be at least 90% of accident design flow rate.

d. At each refueling outage or at any time fire, chemical releases or work done on the filters could alter their integrity or after every 720 hours of charcoal adsorber use since the last test, the following conditions shall be demonstrated before the system can be considered operable:

*Dioctylphthalate Particles

(1) The charcoal shall have a methyl iodine removal efficiency $\geq$ 90% at $\pm$ 20% of the accident design flow rate, 0.05 to 0.15 mg/m^3 inlet methyl iodine concentration, $\geq$ 95% relative humidity and $\geq$ 125°F.

(2) A halogenated hydrocarbon (freon) test on charcoal adsorbers at $\pm$ 20% of the accident design flow rate and ambient conditions shall show $\geq$ 99% halogenated hydrocarbon removal.

(3) A locally generated DOP test of the HEPA filters at $\pm$ 20% of the accident design flow rate and ambient conditions shall show $\geq$ 99% DOP removal.

6. <u>Fuel Storage Building Emergency Ventilation System</u>

a. The fuel storage building emergency ventilation system shall be operated for a minimum of 15 minutes every month when there is irradiated fuel in the spent fuel pit.

b. Prior to handling of irradiated fuel, the following conditions shall be demonstrated before the system can be considered operable:

(1) The pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches of water at ambient conditions and accident design flow rates.

(2) Using either direct or indirect measurements, the flow rate of the system fans shall be shown to be at least 90% of the accident design flow rate.

(3) The filtration system bypass dampers shall be tested to assure proper operation.

c. Prior to handling of irradiated fuel, or at any time fire, chemical releases or work done on the filters could alter their integrity or after every 720 hours of charcoal adsorber use since the last test, the following conditions shall be demonstrated before the system can be considered operable:

(1) Charcoal shall have a methyl iodine removal efficiency $\geq$ 90% at $\pm$ 20% of the accident design flow rate, 0.05 to 0.15 mg/m^3 inlet methyl iodine concentration, $\geq$ 95% relative humidity and $\geq$ 125°F.

(2) A halogenated hydrocarbon (freon) test on charcoal adsorbers at $\pm$ 20% of the accident design flow rate and ambient conditions shall show $\geq$ 99% halogenated hydrocarbon removal.

(3) A locally generated DOP test of the HEPA filters at $\pm$ 20% of the accident design flow rate and ambient conditions shall show $\geq$ 99% DOP removal.

(4) Visual inspection in accordance with ANSI N 510 (1975) of filter installations.

B. Component Tests

1. Pumps

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

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

2. Valves

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

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

c. The following check valves shall be checked for gross leakage every refueling:

857A & G	857J	857S & T	897B
857B	857K	857U & W	897C
857C	857L	895A	897D
857D	857M	895B	838A
857E	857N	895C	838B
857F	857P	895D	838C
857H	857Q & R	897A	838D

d. The following check valves shall be checked for gross leakage midway between refuelings:

838A	895A	897A
838B	895B	897B
838C	895C	897C
838D	895D	897D

Basis

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

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

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

slave relay coil circuits are continuously verified by a built-in monitoring circuit. In addition, the active components (pumps and valves) are to be tested monthly to check the operation of the starting circuits and to verify that the pumps are in satisfactory running order. The test interval of one month is based on the judgment that more frequent testing would not significantly increase the reliability (i.e., the probability that the component would operate when required), yet more frequent testing would result in increased wear over a long period of time.

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

The charcoal portion of the containment air recirculation system is a passive safeguard which is isolated from the cooling air flow during normal reactor operation. Hence, the charcoal should have a long useful lifetime. The filter frames that house the charcoal are stainless steel and should also last indefinitely. However, the visual inspection specified in Section A.4(a) of this specification will be performed to verify that this is, in fact, the case. The iodine removal efficiency cannot be measured with the filter cells in place. Therefore, at periodic intervals a representative sample of charcoal is to be removed and tested to verify that the efficiencies for removal of methyl iodide are obtained.[2] The fuel storage building air treatment system is designed to filter the discharge of the fuel storage building atmosphere to the facility vent during normal conditions. As required by Specifications 3.8.A.11 and 3.8.C.6, the fuel storage building emergency ventilation system must be operable whenever irradiated fuel is being moved. However, if the irradiated fuel has had a continuous 45-day decay period, the fuel storage building emergency ventilation system is not required. The emergency ventilation system is automatically started upon high radiation signal and isolation valves in the ventilation system must close to redirect

air flow through the emergency ventilation system. If these dampers do not operate properly, excessive bypass leakage could occur to negate the usefulness of the HEPA filters and charcoal adosrbers to reduce potential radio-iodine releases to the atmosphere. Therefore, operation of these dampers will be checked before the start of fuel handling operations.

High efficiency particulate absolute (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of these adsorbers for all emergency air treatment systems. The charcoal adsorbers are installed to reduce the potential release of radio-iodine to the environment. The in-place test results should indicate a system leak tightness of less than or equal to one percent leakage for the charcoal adsorbers and a HEPA efficiency of greater than or equal to 99 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a methyl iodide removal efficiency of greater than or equal to 90 percent on the fuel handling system samples, and greater than or equal to 85 percent on the containment system samples for expected accident conditions. With the efficiencies of the HEPA filters and charcoal adsorbers as specified, further assurance is provided that the resulting doses will be less than the 10CFR100 guidelines for the accidents analyzed.

The control room air treatment system is designed to filter the control room atmosphere for intake air and/or for recirculation during control room isolation conditions. The control room air treatment system is designed to automatically start upon control room isolation.

High efficiency particulate absolute (HEPA) filters are installed before the charcoal adsorbers to similarly prevent clogging of these adsorbers. The charcoal adsorbers are installed to reduce the potential intake of radio-iodine by control room personnel. The in-place test results should indicate a system leak tightness of less than or equal to one percent leakage for the charcoal adsorbers and a HEPA filter efficiency of greater than or equal to 99 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a methyl iodide removal efficiency of greater than or equal to 90 percent for expected accident conditions.

With the efficiencies of the HEPA filters and charcoal adsorbers as specified, further assurance is provided that the resulting doses will be less than the allowable levels stated in Criterion 19 of the General Design Criteria for Nuclear Power Plants, Appendix A to 10CFR Part 50.

A pressure drop across the combined HEPA filters and charcoal adsorbers of less than or equal to 6.0 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Pressure drop should be determined at least once per operating cycle to show system performance capability. Proper operation of the system fans should also be verified at least every refueling by either direct or indirect measurements.

If results of charcoal tests are unsatisfactory, two additional samples may be tested. If both of these tests are acceptable, the charcoal may be considered satisfactory for use in the plant. Should the charcoal of any of these air filtration systems fail to satisfy the test criteria outlined in this specification, the charcoal beds will be replaced with new charcoal which satisfies the requirements for new charcoal outlined in Regulatory Guide 1.52 (Revision June, 1973).

The hydrogen recombiner system is an engineered safety feature which would be used only following a loss-of-coolant accident to control the hydrogen evolved in the containment. The system is not expected to be started until approximately 12 days have elapsed following the accident. At this time, the hydrogen concentration in the containment will have reached 2% by volume, which is the design concentration for starting the recombiner system.[3] Actual starting of the system will be based upon containment atmosphere sample analysis. The complete functional tests of each unit at refueling shutdown will demonstrate the proper operation of the recombiner system. More frequent tests of the recombiner control system and air-supply blowers will assure operability of the system. The bi-annual testing of the containment atmosphere sampling system will demonstrate the availability of this system.

For the eight flow distribution valves (856 A, C, D, E, F, H, J and K), verification of the valve mechanical stop adjustments is performed periodically to provide assurance that the high head safety injection flow distribution is in accordance with flow values assumed in the core cooling analysis.

References

(1) FSAR Section 6.2
(2) FSAR Section 6.4
(3) FSAR Section 6.8

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.

Reference

FSAR, Section 8.2

4.7 MAIN STEAM STOP VALVES

Applicability

Applies to periodic testing of the main steam stop valves.

Objective

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

Specification

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

Basis

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

References

(1) FSAR - Section 10.5

(2) FSAR - Section 14.2.5

4.8 AUXILIARY FEEDWATER SYSTEM

Applicability

Applies to periodic testing requirements of the Auxiliary Feedwater System.

Objective

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

Specification

1.a Each auxiliary feedwater pump will be started manually from the control room at monthly intervals with full flow established to the steam generators once every refueling.

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

c Backup supply valves from the city water system will be tested once every refueling.

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

Basis

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

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

Reference

FSAR - Sections 10.4, 14.1.9 and 14.2.5 and response to Question 7.23.

4.9 STEAM GENERATOR TUBE INSERVICE SURVEILLANCE

Applicability

Applies to inservice surveillance of the steam generator tubes.

Objective

To assure the continued integrity of the steam generator tubes that are a part of the primary coolant pressure boundary.

Specification

Steam generator tubes shall be determined operable by the following inspection program and corrective measures:

A. Inspection Requirements

1. Definitions

a. Imperfection is an exception to the dimension, finish, or contour required by drawing or specification.

b. Degradation means a service-induced cracking, wastage, wear or corrosion.

c. Degraded Tube is a tube that contains imperfections caused by degradation large enough to be reliably detected by eddy current inspection. This is considered to be 20% degradation.

d. % Degradation is an estimate % of the tube wall thickness affected or removed by degradation.

e. Defect is an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.

f. Plugging Limit is the imperfection depth at or beyond which the tube must be removed from service. This is considered

to be an imperfection depth of 40%.

g. Tube Inspection is an inspection of tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

2. Sample Size and the Number of Steam Generators to be Inspected.

a. At the first inservice inspection subsequent to the preservice inspection, six percent of the tubes in each of two steam generators shall be inspected as a minimum.

b. At the second inservice inspection subsequent to the preservice inspection, twelve percent of the tubes in one of the two steam generators not inspected during the first inservice inspection shall be inspected as a minimum.

c. At the third inservice inspection subsequent to the preservice inspection, twelve percent of the tubes in the steam generator not inspected during the first two inservice inspections shall be inspected as a minimum.

d. Fourth and subsequent inservice inspections may be limited to one steam generator on a rotating schedule encompassing 3 N% of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances, the sample sequences should be modified to inspect the steam generator with the most severe conditions.

e. Unscheduled inspections should be conducted on the affected steam generator(s) in accordance with the first sample inspection specified in Table 4.9-1 in the event of primary-to-

secondary tube leaks (not including leaks originated from tube-to-tube sheet welds) exceeding technical specifications, a seismic occurrence greater than an operating basis earthquake, a loss-of-coolant accident requiring actuation of engineered safeguards, or a major steam line or feedwater line break.

3. Extent and Result of Steam Generator Tube Inspection

a. The minimum sample size, inspection result classification, and the corresponding action required are specified in Table 4.9-1.

b. Tubes for the inspection should be selected on a random basis except where experience in similar plants with similar water chemistry indicates critical areas to be inspected.

c. The first sample inspection subsequent to the preservice inspection should include all nonplugged tubes that previously had detectable wall penetration (> 20%) and should also include tubes in those areas where experience has indicated potential problems.

d. The second and third sample inspections in Table 4.9-1 may be limited to the partial tube inspection only, concentrating on tubes in the areas of the tube sheet array and on the portion of the tube where tubes with imperfections were found.

e. In all inspections, previously degraded tubes must exhibit significant (> 10%) further wall penetration to be included in the percentage calculation for the result categories in Table 4.9-1.

4. Interval of Inspection

a. The first inservice inspection of steam generators should be performed after six effective full power months but not later than completion of the first refueling outage.

b. Subsequent inservice inspections should be not less than 12 or more than 24 calendar months after the previous inspection.

c. If the results of two consecutive inspections, not including the preservice inspection, all fall in the C-1 category, the frequency of inspection may be extended to 40-month intervals. Also, if it can be demonstrated through two consecutive inspections that previously observed degradation has not continued and no additional degradation has occurred, a 40-month inspection interval may be initiated.

B. Corrective Measures

All leaking tubes and defective tubes should be plugged.

C. Reports

The results of these steam generator tube inservice inspections shall be included in the Annual Operating Report for the period in which this inspection was completed.

Basis

Inservice inspection of steam generators is essential in order to monitor the integrity of the tubing and to maintain surveillance in the event that there is evidence of mechanical damage or progressive

deterioration due to design, manufacturing errors, or chemical imbalance. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

An essentially 100% tube inspection was performed on each tube in every steam generator by eddy current techniques prior to service in order to establish a baseline condition for the tubing. This inspection was conducted under conditions and with equipment and techniques equivalent to those expected to be employed in the subsequent inservice inspections.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those limits found to result in negligible corrosion of the steam generator tubes. If stress corrosion cracking occurs, the extent of cracking during plant operation would be limited by the limitation of steam generator leakage between the primary coolant system and the secondary coolant system. Cracks having a primary-to-secondary leakage less than 500 gallons per day during operation will have an adequate margin of safety against failure due to loads imposed by design basis accidents. Operating plants have demonstrated that primary-to-secondary leakage as low as 0.1 gpm will be detected. Leakage in excess of 500 gallons per day per steam generator or 1 gpm total through all four steam generators will require plant shutdown and an unscheduled eddy current inspection, during which the leaking and defective tubes will be located and plugged. The 500 gallon per day limit is also consistent with the assumptions used to develop the Technical Specification limit for secondary coolant activity.

Wastage-type defects are unlikely with the planned all volatile treatment (AVT) of secondary coolant. However, even if this type of defect occurs, the steam generator tube surveillance specification will identify steam generator tubes with impurifications having a depth greater than 40% of the 0.050 inch tube wall thickness as being unacceptable for continued service. The results of steam generator tube burst and collapse tests have demonstrated that tubes having wall thickness not less than 0.025 inch have adequate margins of safety against failure due to loads imposed by normal plant operation and design basis accidents.

A 10% allowance for tube degradation that may occur between inservice tube examinations added to the 40% tube plugging limit provides an adequate margin to assure that SG tubes acceptable for operation will not have a minimum tube wall thickness less than the acceptable 50% of normal tube wall thickness (i.e., 0.025 in) during the service lifetime of the tubes.

Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect wastage type defects that have penetrated 20% of the original 0.050 inch wall thickness.

TABLE 4.9-1

STEAM GENERATOR TUBE INSPECTION

First Sample Inspection			Second Sample Inspection		Third Sample Inspection	
Minimum Size	Result	Action	Result	Action	Result	Action
S* Tubes per steam generator	C-1					
	C-2	Plug defective tubes. Inspect additional 2S tubes in this SG.	C-1			Go to power.
			C-2	Plug defective tubes. Inspect additional 4 S tubes in this SG.	C-1	
					C-2	Plug defective tubes. Go to power
					C-3	Go to first sample. C-3 action
			C-3	Go to first sample. C-3 action		
	C-3	Inspect all tubes in this SG. Plug defective tubes. Inspect 2 S tubes in each other SC	All other SGs C-1			Go to power
			Some SGs C-2 But no add'l C-3	Go to second sample. C-2 action		
			Add'l SG C-3	Inspect all tubes in all SGs. Plug defective tubes.		Report to NRC. NRC approval req'd prior to startup.

* $S = 3\frac{N}{n}\%$ where N is the number of steam generators in the plant, and n is the number of steam generators inspected during an inspection.

Category C-1: Less than 5% of the total tubes inspected are degraded tubes and none of them is defective.

Category C-2: One or more of the total tubes inspected is defective but not more than 1% of the tubes inspected or between 5 and 10% of the tubes inspected are degraded tubes.

Category C-3: More than 10% of the total tubes inspected are degraded or more than 1% of the tubes inspected are defective.

4.10 SEISMIC INSTRUMENTATION

Applicability

Applies to testing of seismic monitoring instruments.

Objective

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

Specification

4.10.1 Each of the seismic monitoring instruments in Table 4.10-1 shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.10-2.

4.10.2 If the number of OPERABLE seismic monitoring instruments is less than that required by Table 4.10-1, restore the inoperable instrument(s) to OPERABLE status within 30 days.

4.10.3 With one or more seismic monitoring instruments inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrument(s) to OPERABLE status.

4.10.4 Each of the seismic monitoring instruments actuated during a seismic event shall be restored to OPERABLE status and a CHANNEL CALIBRATION performed within 48 hours following the seismic event. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days describing the magnitude, frequency spectrum and resultant effect upon facility features important to safety.

Basis

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility and is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes", April, 1974.

TABLE 4.10-1

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE
1. Triaxial Time-History Accelographs		
a. EL 46'-0" VC Base Mat	0 to ± 1G	1*
b. EL 99'-0" VC Wall	0 to ± 1G	1*
2. Triaxial Peak Accelographs		
a. STM GEN # 31	0 to ± 2G	1
b. RC Pump # 31	0 to ± 2G	1
c. Pressurizer	0 to ± 2G	1
3. Triaxial Response-Spectrum Recorders		
a. EL 46'-0" VC Base Mat	0 to ± 1G	1*

* With reactor control room indication

TABLE 4.10-2

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATION	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. Triaxial Time-History Accelographs			
a. EL 46'-0" VC Base Mat	M*	R	SA
b. EL 99'-0" VC Wall	M*	R	SA
2. Triaxial Peak Accelographs			
a. STM GEN #31	NA	R	NA
b. RC Pump #31	NA	R	NA
c. Pressurizer	NA	R	NA
3. Triaxial Response-Spectrum Recorders			
a. EL 46'-0" VC Base Mat **	M	R	SA

* Except seismic trigger
** With reactor control room indications.

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

2. The average enrichment of the initial core is a nominal 2.8 weight per cent of U-235. Three fuel enrichments are used in the initial core. The highest enrichment is a nominal 3.3 weight per cent of U-235.(2)

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

4. Burnable poison rods are incorporated in the initial core. There are 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.

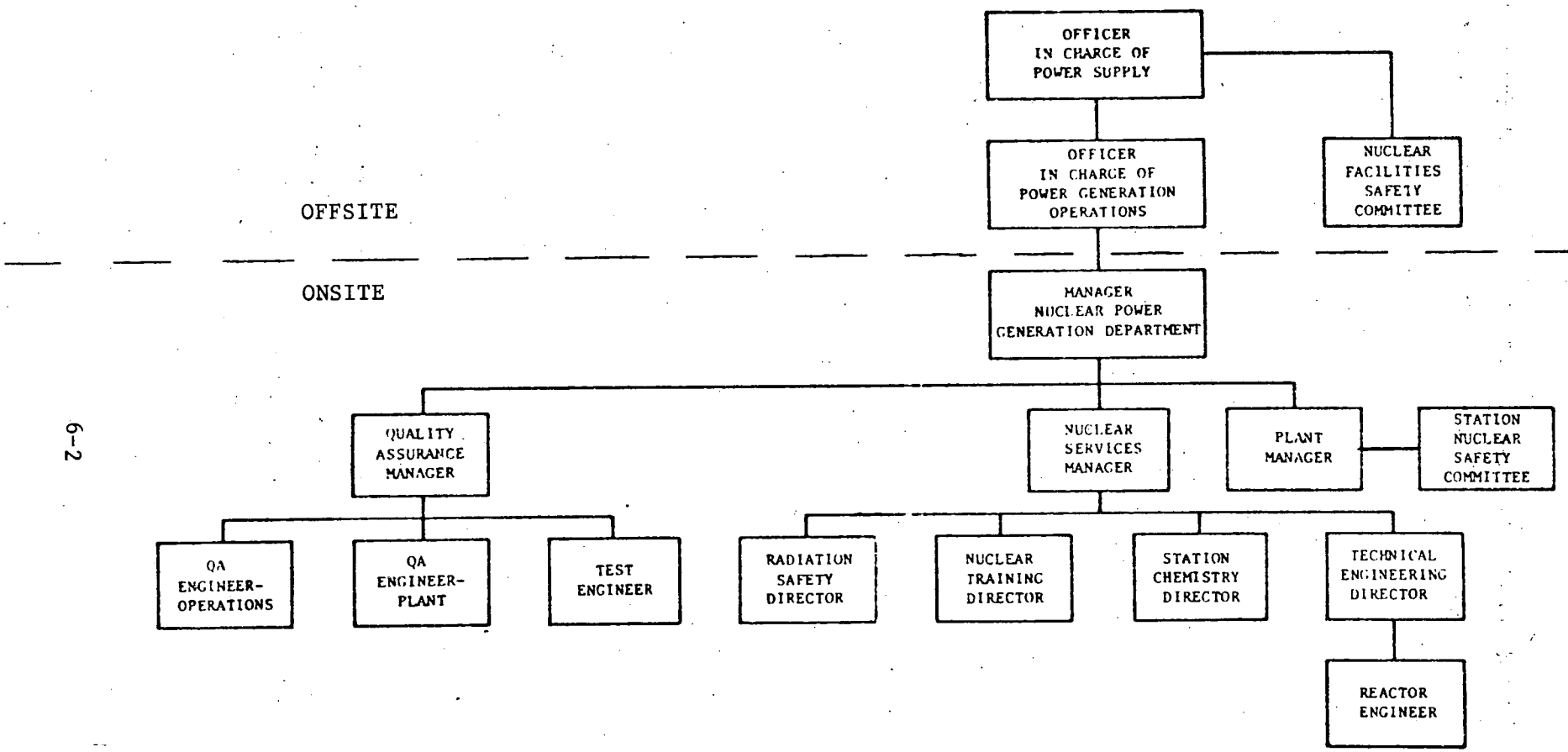


Figure 6.2-1 Facility Management and Technical Support Organization

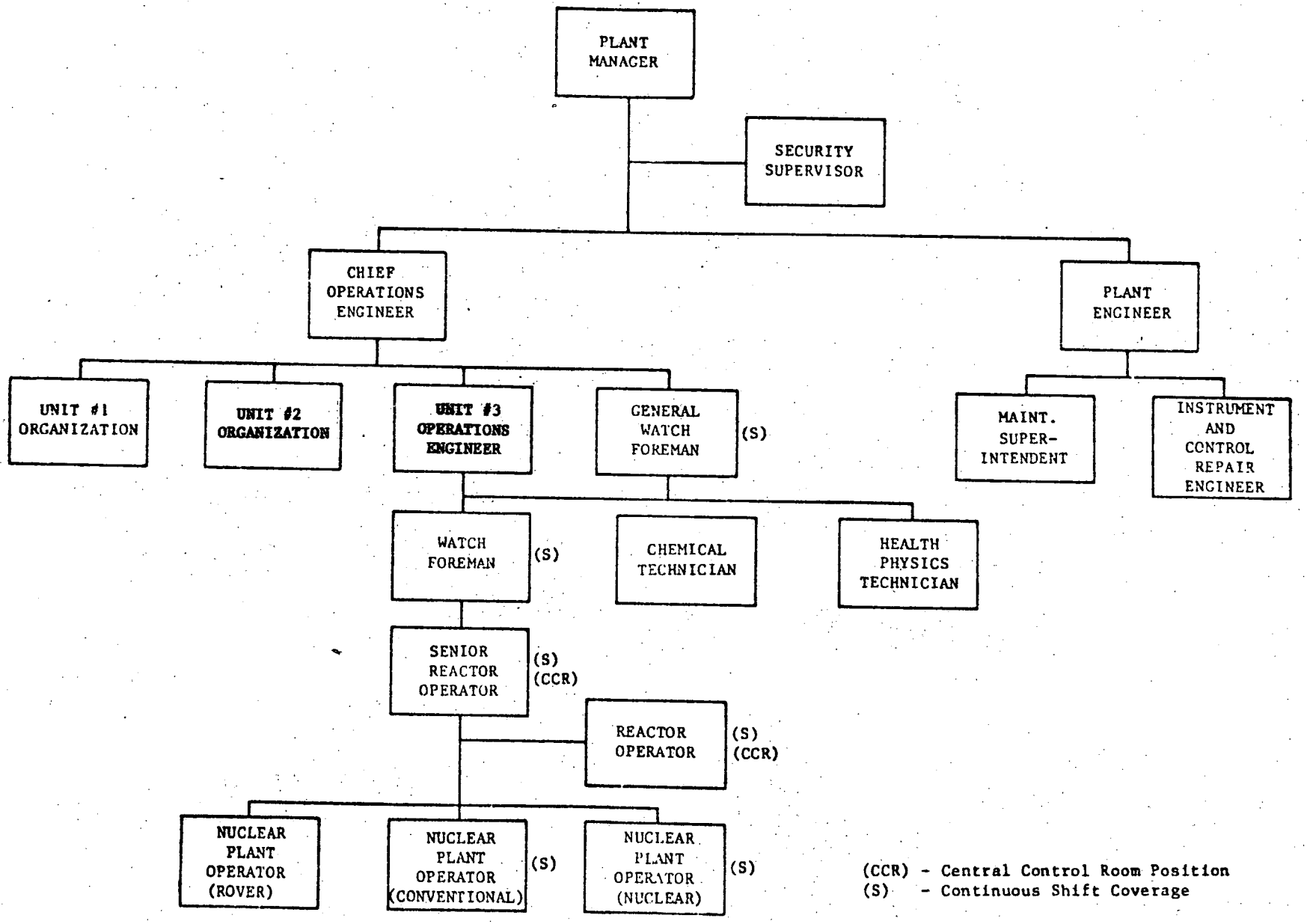


Figure 6.2-2 Facility Organization

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.

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 have voting rights.

Consultants

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

Meeting Frequency

6.5.2.5 The NFSC shall meet at least once per calendar quarter 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. REPORTABLE 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.

Audits

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

a. The conformance of facility operation to all provisions contained within the 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 REPORTABLE OCCURENCE ACTION

6.6.1 The following actions shall be taken in the event of a REPORTABLE OCCURRENCE:

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

b. Each Reportable Occurrence Report submitted to the Commission 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 (November 1972) 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 Reportable Occurence 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 4 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 license 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.

b. Sealed source leakage on excess of limits (Specification 3.9)

c. Inoperable seismic monitoring instrumentation (Specification 4.10)

d. Seismic event analysis (Specification 4.10)

e. Primary coolant activity on excess of limits (Specification 3.1.D)

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. REPORTABLE 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 any 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 20.103(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 protection 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 3)

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/NATIONAL INSTITUTE FOR OCCUPATIONAL SAFETY AND HEALTH APPROVALS
I. AIR-PURIFYING RESPIRATORS			
Facepiece, half-mask(4),(7)	NP	5	30 CFR Part 11 Subpart K
Facepiece, full(7)	NP	100	30 CFR Part 11 Subpart K
II. ATMOSPHERE-SUPPLYING RESPIRATOR			
1. Airline respirator			
Facepiece, half-mask	CF	100	30 CFR Part 11 Subpart J
Facepiece, full	CF	1,000	30 CFR Part 11 Subpart J
Facepiece, full(7)	D	100	30 CFR Part 11 Subpart J
Facepiece, full	PD	1,000	30 CFR Part 11 Subpart J
Hood	CF	(5)	30 CFR Part 11 Subpart J
Suit	CF	(5)	(6)
2. Self-contained breathing apparatus (SCBA)			
Facepiece, full(7)	D	100	30 CFR Part 11 Subpart H
Facepiece, full	PD	1,000	30 CFR Part 11 Subpart H
Facepiece, full	R	100	30 CFR Part 11 Subpart H
III. COMBINATION RESPIRATOR Any combination of air-purifying and atmosphere-supplying respirator		Protection factor for type and mode of operation as listed above	30 CFR Part 11 § 11.63(b)

TABLE NOTATION

[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.

[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.

TABLE 6.12-1 (Page 3 of 3)

TABLE NOTATION

[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 current 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 or the National Institute for Occupational Safety and Health 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 or the National Institute for Occupational Safety and Health in accordance with its applicable schedules.

NOTE 2: Radioactive contaminants for which the concentration values in Appendix B, Table I 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.