CONSUMERS POWER COMPANY

PALISADES PLANT

TECHNICAL SPECIFICATIONS

APPENDIX A

TO

PROVISIONAL OPERATING LICENSE DPR-20

TECHNICAL SPECIFICATIONS

FOR THE

PALISADES PLANT

CONSUMERS POWER COMPANY

DOCKET NO 50-255

ERRATA SHEET NO. 1

For Appendix A to Provisional Operating License DPR-20
Technical Specifications for the Palisades Plant,
Consumers Power Company, Docket No. 50-255

pg. i, item 3.9	- Change page number from 3-43 to 3-44.
pg. 3-8a, Fig. 3-1	- Delete notation "Rev 1/70".
pg. 3-8b, Fig. 3-2	- Delete notation "Rev 1/70".
pg. 3-13, line 15	- Change, "...2 meters period second..." to "...2 meters per second...".
pg. 3-14, line 15	- Change, "...measurement". to "...measurement of $\bar{\bar{E}}$".
pg. 3-44, lines 23, 27, & 28	- Change "activity" to "radioactivity".
pg. 3-52, Spec. 3.10.4 b	- Change to, "A control rod is considered inoperable if it cannot be moved by its operator or if it cannot be tripped. A part-length rod is considered inoperable if it is not fully withdrawn from the core and cannot be moved by its operator. If more than one control rod or part-length rod becomes misaligned or inoperable, the reactor shall be placed in the hot shutdown condition within 12 hours."
pg. 3-56, under Bases item d	- Change to, "Confirm that readings from the out-of-core split detectors are as expected and...".
pg. 4-12, footnote P	- Add "...if not done previous week".
pg. 4-14, footnote (3)	- Change, "...Specification 3.1.5 (b),..." to "...Specification 3.1.5,...".
pg. 4-22a, Table 4.3.3	- Change heading of first column to "Scheduled Refueling Number". Change heading of third column to "Estimated Target Fast Neutron Fluence - nvt, (N/cm^2)".

pg. 4-22a, Table 4.3.3	- Add to heading of fourth column the units "(MWD thermal)". Change the last entry under: the first column from 40 to 39, the third column from 3.4×10^{19} to 3.3×10^{19} and the fourth column from 3.0×10^{7} to 2.93×10^{7}.
pg. 4-41, line 15	- Change "...paralled..." to "...paralleled...".
pg. 4-47a, line 3	- Delete the word "public".

PALISADES PLANT
TECHNICAL SPECIFICATIONS

TABLE OF CONTENTS

TABLE OF CONTENTS (Contd)

TECHNICAL SPECIFICATIONS

1.0 DEFINITIONS

The following terms are defined for uniform interpretation of these Technical Specifications:

1.1 REACTOR OPERATING CONDITIONS

Rated Power

A steady state reactor core output of 2200 MWt.

Reactor Critical

The reactor is considered critical for purposes of administrative control when the neutron flux logarithmitic range channel instrumentation indicates greater than 10^{-4}% of rated power.

Power Operation Condition

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

Hot Standby Condition

The reactor is considered to be in a hot standby condition if the average temperature of the primary coolant (T_{avg}) is greater than 525°F and any of the control rods are withdrawn and the neutron flux power range instrumentation indicates less than 2% of rated power.

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 greater than 525°F.

Refueling Shutdown Condition

When the primary coolant is at refueling boron concentration and T_{avg} is less than 210°F.

Cold Shutdown Condition

When the primary coolant is at shutdown boron concentration and T_{avg} is less than 210°F.

Refueling Operation

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

1.1 REACTOR OPERATING CONDITIONS (Contd)

Shutdown Boron Concentration

Boron concentration sufficient to provide $k_{eff} \leq 0.98$ with all control rods in the core and the highest worth control rod fully withdrawn.

Refueling Boron Concentration

Boron concentration of coolant at least 1720 ppm (corresponding to a shutdown margin of at least 5% $\Delta\rho$ with all control rods withdrawn).

Quadrant Power Tilt

The difference between nuclear power in any core quadrant and the average in all quadrants.

1.2 PROTECTIVE SYSTEMS

Instrument Channels

One of four independent measurement channels, complete with the sensors, sensor power supply units, amplifiers, and bistable modules provided for each safety parameter.

Reactor Trip

The de-energizing of the control rod drive mechanism (CRDM) magnetic clutch holding coils which releases the control rods and allows them to drop into the core.

Reactor Protective System Logic

This system utilizes relay contact outputs from individual instrument channels to provide the reactor trip signal for de-energizing the magnetic clutch power supplies. The logic system is wired to provide a reactor trip on a 2-of-4 or 2-of-3 basis for any given input parameter.

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.

Engineered Safety Features System Logic

This system utilizes relay contact outputs from individual instrument channels to provide a dual channel (right and left) signal to initiate independently the actuation of engineered safety feature equipment connected to diesel generator 1-2 (right channel) and diesel generator 1-1 (left channel). The logic system is wired to provide an appropriate signal for the actuation of the engineered safety feature equipment on a 2-of-4 basis for any given input parameter.

1.3 INSTRUMENTATION SURVEILLANCE

Channel Check

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

Channel Functional Test

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

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 equipment action, alarm, interlocks or trip, and shall be deemed to include the channel functional test.

1.4 MISCELLANEOUS DEFINITIONS

Operable

A system or component is operable if it is capable of fulfilling its design functions.

Operating

A system or component is operating if it is performing its design functions.

Control Rods

All full-length shutdown and regulating rods.

Containment Integrity

Containment integrity is defined to exist when all of the following are true:

a. All nonautomatic containment isolation valves and blind flanges are closed.

b. The equipment door is properly closed and sealed.

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

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

e. The uncontrolled containment leakage satisfies Specification 4.5.1.

1.4 MISCELLANEOUS DEFINITIONS (Contd)

Abnormal Occurrence

An abnormal occurrence is defined as any of the following:

a. Actual safety system settings less conservative than limiting safety settings specified in these technical specifications.

b. The occurrence of any plant condition violating a limiting condition of operation.

c. Abnormal degradation of one of the several boundaries designed to contain the radioactive materials resulting from the fission process.

d. Occurrences which could, or threaten to, render an engineered safety system or reactor protective system incapable of performing its intended safety function.

e. Uncontrolled or unanticipated change in reactivity greater than 1% $\Delta\rho$.

f. Inadequacies in the implementation of administrative or procedural controls, such that the inadequacy could have caused the existence or development of an unsafe condition.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS - REACTOR CORE

Applicability

This specification applies to the limiting combinations of reactor power, primary coolant system flow, temperature and pressure during operation.

Objective

To maintain the integrity of the fuel cladding and prevent the release of significant amounts of fission products to the primary coolant.

Specifications

The reactor power level shall not exceed the allowable limit for the pressurizer pressure and the cold leg temperatures (shown in Figures 2-1, 2-2, and 2-3) for 2-, 3- and 4-pump operation.

The safety limit is exceeded if the point defined by the combination of primary coolant cold leg temperature and power level is at any time above the appropriate pressurizer 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 normal operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is large enough so that the clad surface temperature is only slightly greater than the coolant temperature. The upper boundary of the nucleate boiling regime is termed "departure from nucleate boiling" (DNB). At this point, there is a sharp reduction of the heat transfer coefficient, which would result in high-cladding temperatures and the possibility of cladding failure. Although DNB is not an observable parameter during reactor operation, the observable parameters of thermal power, primary coolant flow, temperature and pressure, can be related to DNB through the use of the "W-3 DNB Correlation."[1] The W-3 DNB Correlation has been developed to predict DNB and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual 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 1.3. A DNBR of 1.3 corresponds to a 95%

2.1 SAFETY LIMITS - REACTOR CORE (Contd)

probability at a 95% confidence level that DNB will not occur which is considered an appropriate margin to DNB for all operating conditions.[1] The curves of Figures 2-1, 2-2, and 2-3 represent the loci of points of thermal power, primary coolant system pressure and average temperature of various pump combinations for which the DNBR is 1.3. The area of safe operation is below these lines. For 3- and 2-pump operation, the limiting condition is void fraction rather than DNBR. The void fraction limits assure stable flow and maintenance of DNBR greater than 1.3.

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

$$F_q^N = 3.62 \text{ and } F_{\Delta H}^N = 1.94$$

These limiting hot channel factors are higher than those calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion. (Control rod insertion limits are covered in Specification 3.10.) Somewhat worse hot channel factors could occur at lower power levels because additional control rods may be in the core; however, the control rod insertion limits dictated by Figure 3-6 insure that the minimum DNBR is always greater at part-power than at full-power.

Flow maldistribution effects of operation under less than full primary coolant flow have been evaluated via model tests.[2] The flow model data established the maldistribution factors and hot channel inlet temperatures for the thermal analyses that were used to establish the safe operating envelopes presented in Figures 2-1 and 2-2. These figures were established on the basis that the thermal margin for part-loop operation should be equal to or greater than the thermal margin for normal operation.

The reactor protective system is designed to prevent any anticipated combination of transient conditions for primary coolant system temperature, pressure and thermal power level that would result in a DNBR of less than 1.3.[3]

References

(1) FSAR, Section 3.3.3.5.

(2) FSAR, Section 3.3.3.3, Appendix C

(3) FSAR, Section 14.1.

2.2 SAFETY LIMITS - PRIMARY COOLANT SYSTEM PRESSURE

Applicability

Applies to the limit on primary coolant system pressure.

Objective

To maintain the integrity of the primary coolant system and to prevent the release of significant amounts of fission product activity to the primary coolant.

Specification

The primary coolant system pressure shall not exceed 2750 psia when there are fuel assemblies in the reactor vessel.

Basis

The primary coolant system[1] serves as a barrier to prevent radionuclides in the primary coolant from reaching the atmosphere. In the event of a fuel cladding failure, the primary coolant system is the foremost barrier against the release of fission products. Establishing a system pressure limit helps to assure the continued integrity of both the primary coolant system and the fuel cladding. The maximum transient pressure allowable in the primary coolant system pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient presssure allowable in the primary coolant system piping, valves and fittings under ASA Section B31.1 is 120% of design pressure. Thus, the safety limit of 2750 psia (110% of the 2500 psia design pressure) has been established.[2] The settings and capacity of the secondary coolant system safety valves (985-1025 psia),[3] the reactor high-pressure trip (2400 psia) and the primary safety valves (2500-2580 psia)[4] have been established to assure never reaching the primary coolant system pressure safety limit. The initial hydrostatic test was conducted at 3125 psia (125% of design pressure) to verify the integrity of the primary coolant system. Additional assurance that the nuclear steam supply system (NSSS) pressure does not exceed the safety limit is provided by setting the pressurizer power-operated relief valves at 2400 psia and the secondary coolant system steam-dump and bypass valves at 900 psia.

References

(1) FSAR, Section 4.

(2) FSAR, Section 4.3.

(3) FSAR, Section 4.3.4.

(4) FSAR, Section 4.3.9.

2.3 LIMITING SAFETY SYSTEM SETTINGS - REACTOR PROTECTIVE SYSTEM

Applicability

This specification applies to reactor trip settings and bypasses for instrument channels.

Objective

To provide for automatic protective action in the event that the principal process variables approach a safety limit.

Specification

The reactor protective system trip setting limits and the permissible bypasses for the instrument channels shall be as stated in Table 2.3.1.

Table 2.3.1

Reactor Protective System Trip Setting Limits

	Four Primary Coolant Pumps Operating	Three Primary Coolant Pumps Operating	Two Primary Coolant Pumps Operating
1. High Power Level (1)	$\leq$ 106.5% of Rated Power	$\leq$ 45% of Rated Power	$\leq$ 25% of Rated Power
2. Low Primary Coolant Flow (2)	$\geq$ 95% of Primary Coolant Flow With 4 Pumps Operating	$\geq$ 71% of Primary Coolant Flow With 4 Pumps Operating	$\geq$ 46% of Primary Coolant Flow With 4 Pumps Operating
3. High Pressurizer Pressure	$\leq$ 2400 Psia	$\leq$ 2400 Psia	$\leq$ 2400 Psia
4. Thermal Margin/Low Pressure (2)(3)	$P_T \geq$ Applicable Limits To Satisfy Figure 2-3	Replaced by High-Power Level Trip and 1750 Psia Minimum Low-Pressure Setting	Replaced by High-Power Level Trip and 1750 Psia Minimum Low-Pressure Setting
5. Low Steam Generator Water Level	At the Center Line of Feed-Water Ring (6'-0" Below Normal Water Level)	At the Center Line of Feed-Water Ring (6'-0" Below Normal Water Level)	At the Center Line of Feed-Water Ring (6'-0" Below Normal Water Level)
6. Low Steam Generator Pressure (2)	$\geq$ 500 Psia	$\geq$ 500 Psia	$\geq$ 500 Psia
7. Containment High Pressure	$\leq$ 5 Psig	$\leq$ 5 Psig	$\leq$ 5 Psig

(1) Below 5% rated power, the trip setting may be manually reduced by a factor of 10.
(2) May be bypassed below 10-4% of rated power provided auto bypass removal circuitry is operable.
(3) T_h and T_c in ^{o}F. Minimum trip setting shall be 1750 psia for all pump combinations.

2.3 LIMITING SAFETY SYSTEM SETTINGS - REACTOR PROTECTIVE SYSTEM (Contd)

Basis

The reactor protective system consists of four instrument channels to monitor selected plant conditions which will cause a reactor trip if any of these conditions deviate from a preselected operating range to the degree that a safety limit may be reached.

1. High Power Level - A reactor trip at high power level (neutron flux) is provided to prevent damage to the fuel cladding resulting from some reactivity excursions too rapid to be detected by pressure and temperature measurements.

   During normal plant operation with all primary coolant pumps operating, reactor trip is initiated when the reactor power level reaches 106.5% of indicated full power. Adding to this the possible variation in trip point due to calibration and instrument errors, the maximum actual steady-state power at which a trip would be actuated is 112%, which was used for the purpose of safety analysis.(1)

   Provisions have been made to select different high-power level trip points for various combinations of primary coolant pump operation as described below under "Low Primary Coolant Flow."(2)

   If reactor operation at less than 10% of full power is required for an extended period of time, provisions have been made to allow the operator to decrease the indicated power range by a factor of 10, which will also decrease the high-power level trip point by a factor of 10 to 10.65% of indicated full power.(2) Administrative procedures will allow this range change to be made during reactor start-up and also between 5% and 8% of rated power when the reactor power is reduced to that level.

2. Low Primary Coolant Flow - A reactor trip is provided to protect the core against DNB should the coolant flow suddenly decrease significantly. Provisions are made in the reactor protective system to permit operation of the reactor at reduced power if one or two coolant pumps are taken out of service. These low-flow and high-flux settings have been derived in consideration of instrument errors and response times of equipment involved to maintain the DNBR above 1.3 under normal operation(5) and expected transients.(4) For reactor operation with one or two coolant pumps inoperative, the low-flow

trip points and the overpower trip points must be manually changed to the specified values for the selected pump condition by means of a set point selector switch. Flow in each of the four coolant loops is determined from a measurement of pressure drop from inlet to outlet of the steam generators. The total flow through the reactor core is measured by summing the loop pressure drops across the steam generators and correlating this pressure sum with the pump calibration flow curves.

The percent of normal core flow is shown in the following table:[5]

4 Pumps	100.0%
3 Pumps	74.7%
2 Pumps	48.7%

During four-pump operation, the low-flow trip setting of 95% insures that the reactor cannot operate when the flow rate is less than 93% of the nominal value considering instrument errors.[3] The high-power level trip and the low primary coolant flow trip are reduced to compensate for the corresponding core flow reduction experienced with fewer than four pumps in operation. The trip points are shown in Table 2.3.1.

3. High Pressurizer Pressure - A reactor trip for high pressurizer pressure is provided in conjunction with the primary and secondary safety valves to prevent primary system overpressure (Specification 3.1.7). In the event of loss of load without reactor trip, the temperature and pressure of the primary coolant system would increase due to the reduction in the heat removed from the coolant via the steam generators. The power-operated relief valves are set to operate concurrently with the high pressurizer pressure reactor trip. This setting is 100 psi below the nominal safety valve setting (2500 psia) to avoid unnecessary operation of the safety valves. This setting is consistent with the trip point assumed in the accident analysis.[1]

4. Thermal Margin/Low-Pressure Trip - A reactor trip is provided to prevent operation when the DNBR is less than 1.3, or a void fraction limit which could result in local flow instability is exceeded.[6] The thermal and hydraulic safety limits shown on Figures 2-1, 2-2 and 2-3 for two-, three- and four-primary coolant pump operation,

2.3 LIMITING SAFETY SYSTEM SETTINGS - REACTOR PROTECTIVE SYSTEM (Contd)

respectively, define the limiting values of primary coolant pressure, reactor inlet temperature, and reactor power level in order that the thermal criteria given in Reference 6 are not exceeded. Figure 3-31 of the FSAR shows the lines of constant DNBR and forms the bases of the thermal limit curves shown on Figures 2-1, 2-2 and 2-3.

To avoid the possibility of DNBR resulting from local flow oscillations, a conservative limit to prevent flow instabilities has been established. A thermal margin trip (TM trip) will occur before the flow instability limit is reached; thus, DNBR resulting from flow oscillations is prevented.[7]

The corollary thermal and hydraulic design bases for the settings are set forth in Section 3 of the FSAR. The low-pressure trip (LP) of 1750 psia also causes a reactor trip in the unlikely event of a loss-of-coolant accident.

A TM trip is initiated by a continuously computed function of reactor inlet temperature, reactor outlet temperature and pressure. The generated function represents a measure of the combination of pressure, temperature and power to flow ratio (as indicated by reactor temperature rise) at which a shutdown should occur to prevent violating the DNBR and void fraction limits. The formula for the TM trip (P_T) set point, $P_T = AT_h - BT_c - C$, is based on reactor inlet temperature (T_c) and reactor outlet temperature (T_h), which defines the minimum allowable pressure for operation and is continuously compared to the measured pressurizer pressure. Conservative values for these constants (A, B and C) will be determined during the power test program at power levels below those where TM protection would be required. These settings will be such that, including instrumentation errors, operation in violation of the thermal limits as shown on Figure 2-3 will result in a reactor trip. At no time will the calculated trip pressure (P_T) be less than the minimum allowable pressure for the applicable reactor condition shown on Figure 2-3.

For operations with either two or three primary coolant pumps, the high-power level trip will be reduced such that operations in violation

2.3 LIMITING SAFETY SYSTEM SETTINGS - REACTOR PROTECTIVE SYSTEM (Contd)

of the core safety limits as shown on Figure 2-1 or 2-2, respectively, are not possible. These reduced high-power level trip settings and the TM trip setting will be reviewed after the power test program has been completed and the results evaluated.

In order to compensate for maximum temperature and pressure measurement errors, the values chosen will be such that a TM trip will always occur before the thermal limits are exceeded. Accordingly, the maximum error assumed is -139 psi[8] in the accident analysis. The trip pressure (P_T) calculated by the TM function should, therefore, be at least 139 psi above the pressure shown on Figure 2-3 for a given combination of reactor inlet temperature and pressure. For four-pump operation, the trip pressure (P_T) is continuously indicated and can be compared to the thermal limit curve to check for proper adjustment in operation of the TM trip circuit. The safety setting limits used when operating at either 45% of full power for three-pump operation or at 25% of full power for two-pump operation assure that with the secondary coolant system safety valves set at 1000 psia, and with a fixed power level, the T_c cannot be above the limiting T_c temperature. Therefore, with a reactor trip occurring at 1750 psia (which sets a minimum pressure condition that cannot be less than an acceptable pressure level), it is impossible for the core DNBR and/or void fraction safety limits to be violated. These lower three- and two-pump power levels will be set by the high-power level - low-primary coolant flow switch.

5. Low Steam-Generation-Water-Level - The low steam-generation-water-level reactor trip protects against the loss of feedwater flow accidents and assures that the design pressure of the primary coolant system will not be exceeded. The specified set point assures that there will be sufficient water inventory in the steam-generator at the time of trip to provide a 15-minute margin before the auxiliary feedwater is required.[9]

   The setting listed in Table 2.3.1 assures that the heat transfer surface (tubes) is covered with water when the reactor is critical.

2.3 LIMITING SAFETY SYSTEM SETTINGS - REACTOR PROTECTIVE SYSTEM (Contd)

6. Low Steam-Generator Pressure - A reactor trip on low steam-generator secondary pressure is provided to protect against an excessive rate of heat extraction from the steam-generators and subsequent cooldown of the primary coolant. The setting of 500 psia is sufficiently below the full-load operating point of 739 psia so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This setting was used in the accident analysis.(8)

7. Containment High Pressure - A reactor trip on containment high pressure is provided to assure that the reactor is shut down upon the initiation of the safety injection system. The setting of this trip is identical to that of the containment high-pressure safety injection signal.(10)

References

(1) FSAR, Section 14.1.

(2) FSAR, Section 7.2.3.2.

(3) FSAR, Section 7.2.3.3.

(4) FSAR, Section 14.7.4.

(5) FSAR, Section 3.3.3.

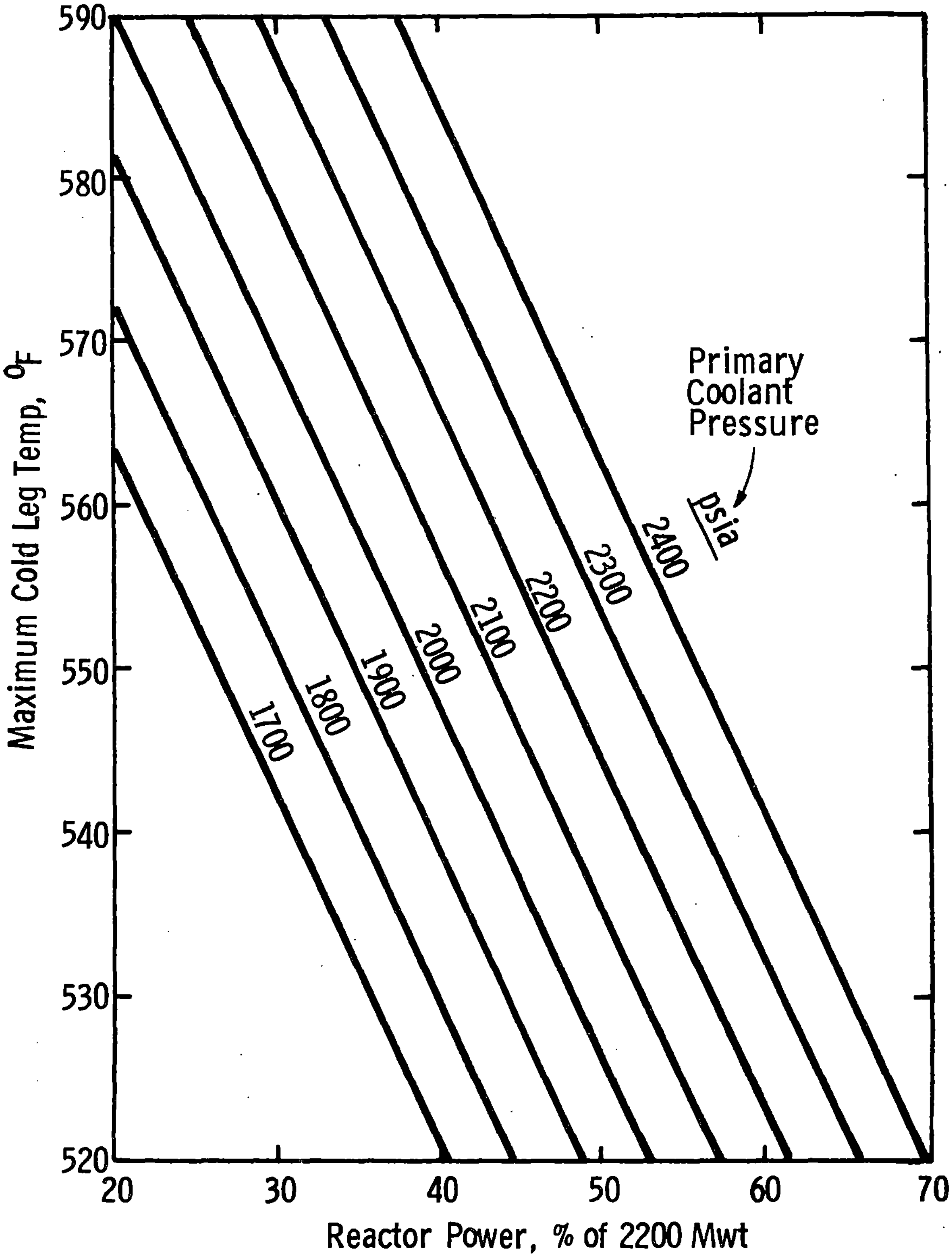
(6) FSAR, Section 3.3.3.5.

(7) FSAR, Section 3.3.3.6.

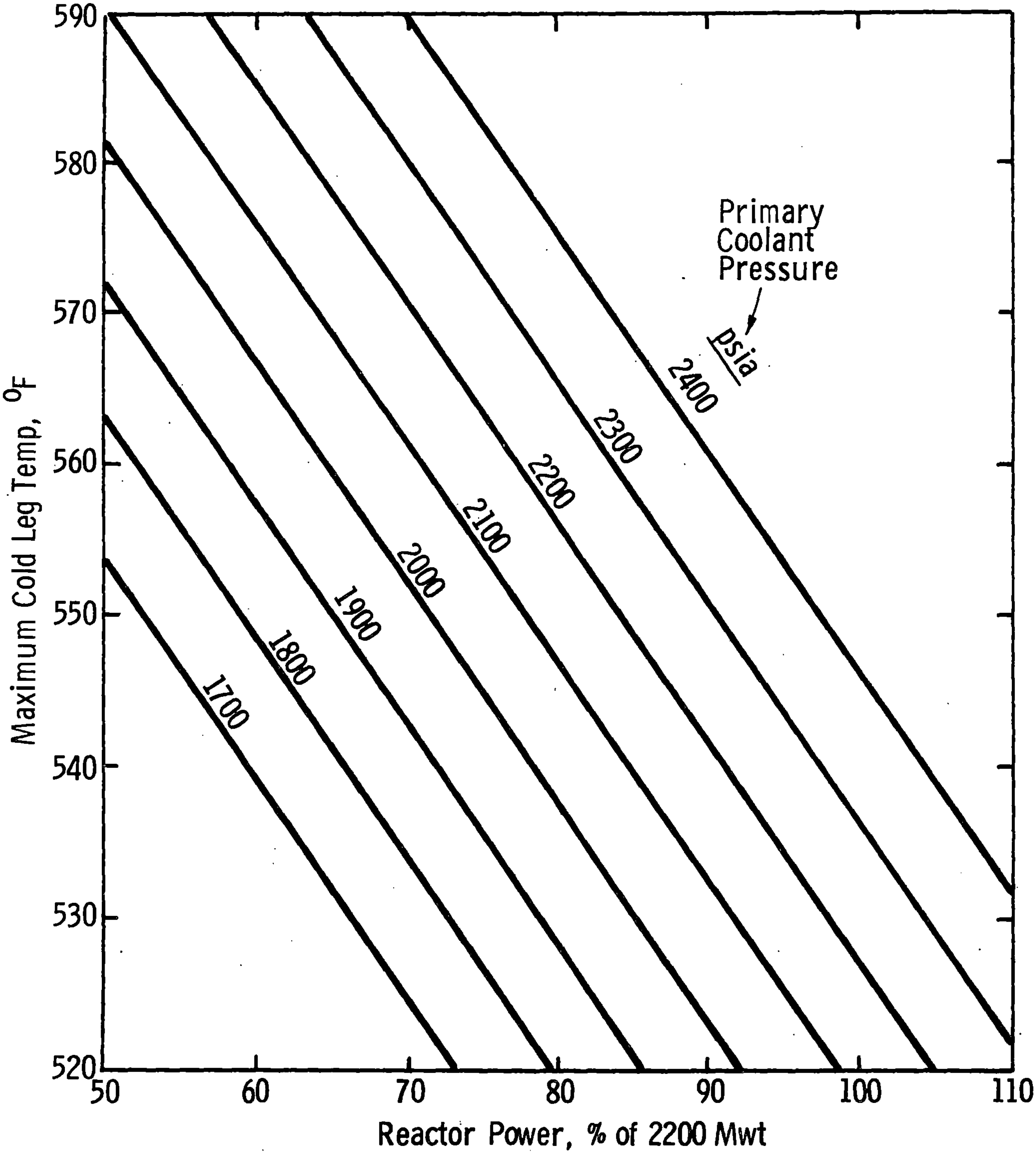
(8) FSAR, Section 14.14.3.

(9) FSAR, Section 14.13.3.

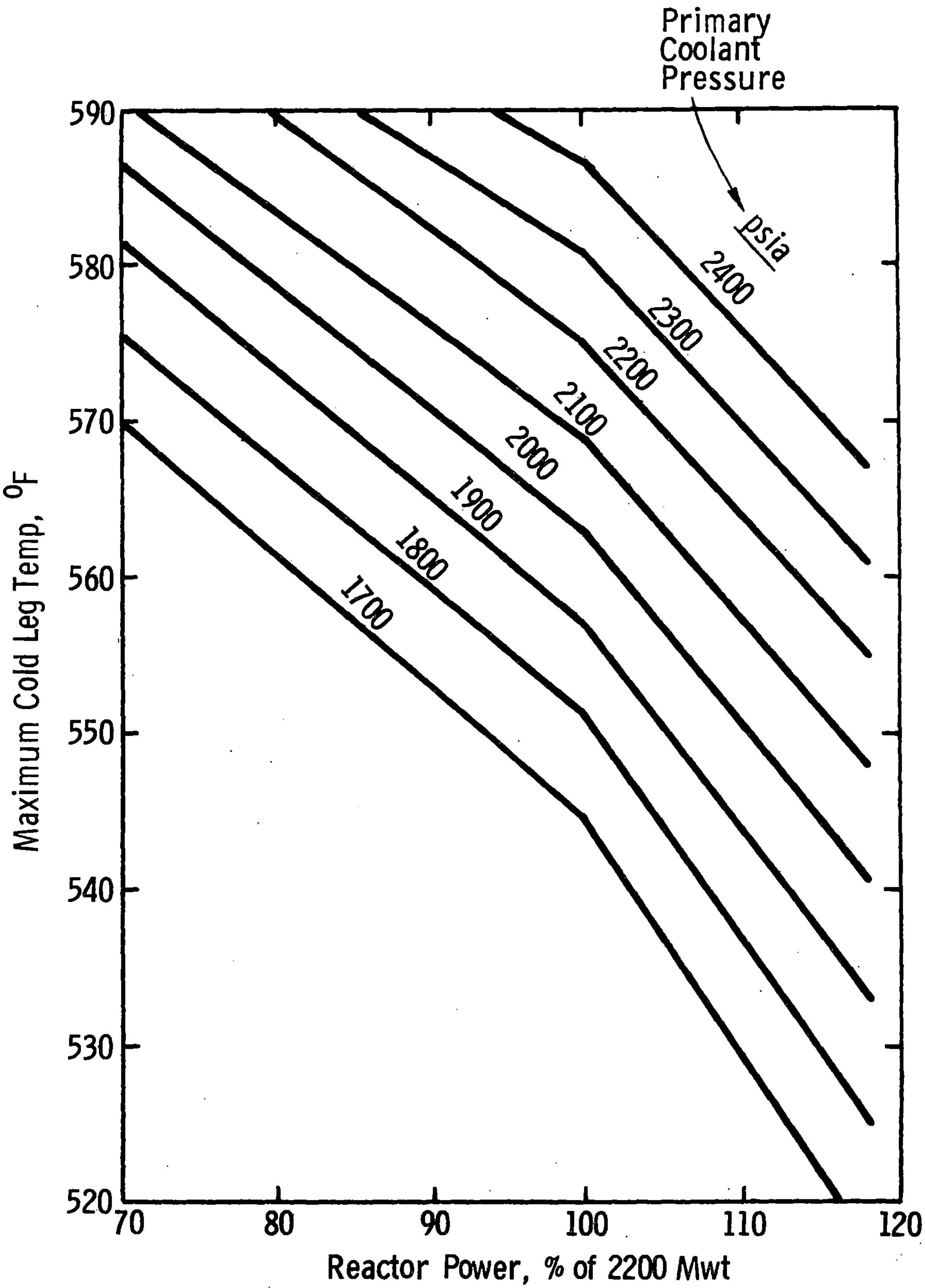
(10) FSAR, Amendment No 17, Item 4.0



Reactor Core Safety Limits 2 Pump Operation	Palisades Technical Specifications	Figure 2-1



Reactor Core Safety Limits 3 Pump Operation	Palisades Technical Specifications	Figure 2-2



Reactor Core Safety Limits 4 Pump Operation	Palisades Technical Specifications	Figure 2-3

3.0 LIMITING CONDITIONS FOR OPERATION

3.1 PRIMARY COOLANT SYSTEM

Applicability

Applies to the operable status of the primary coolant system.

Objective

To specify certain conditions of the primary coolant system which must be met to assure safe reactor operation.

Specifications

3.1.1 Operable Components

a. At least one primary coolant pump or one shutdown cooling pump shall be in operation whenever a change is being made in the boron concentration of the primary coolant.

b. Except for special tests during initial start-up testing, at least two primary coolant pumps shall be in operation whenever the reactor is critical.

c. Minimum flow conditions for various reactor power levels shall be maintained as shown in Table 2.3.1.

d. Both steam generators shall be capable of performing their heat transfer function whenever the average temperature of the primary coolant is above 325°F.

e. Maximum primary system pressure differentials shall not exceed the following:

(1) Maximum steam generator operating transient differential of 1980 psi.

(2) Maximum hydrostatic test differential pressure of 3125 psia. A maximum of 10 cycles of hydrostatic pressure differential are allowed.

(3) Primary side leak tests shall be conducted at normal operating pressure and temperature <400°F. If conducted above 212°F, the secondary side shall be filled to within the normal operating range and pressurized to a pressure corresponding to saturated pressure of the primary fluid temperature.

3.1 PRIMARY COOLANT SYSTEM (Contd)

(4) Maximum secondary hydrostatic test differential shall not exceed 1250 psia. A minimum temperature of $100^{\circ}F$ is required. Only 10 cycles are permitted.

(5) Maximum secondary leak test pressure shall not exceed 1000 psia. A minimum temperature of $100^{\circ}F$ is required.

Basis

When primary coolant boron concentration is being changed, the process must be uniform throughout the primary coolant system volume to prevent stratification of primary coolant at lower boron concentration which could result in a reactivity insertion. Sufficient mixing of the primary coolant is assured if one shutdown cooling or one primary coolant pump is in operation.(1) The shutdown cooling pump will circulate the primary system volume in less than 60 minutes when operated at rated capacity. The pressurizer volume is relatively inactive, therefore will tend to have a boron concentration higher than rest of the primary coolant system during a dilution operation. Administrative procedures will provide for use of pressurizer sprays to maintain a nominal spread between the boron concentration in the pressurizer and the primary system during the addition of boron.(2)

Both steam generators are required to be operable whenever the temperature of the primary coolant is greater than the design temperature of the shutdown cooling system to assure a redundant heat removal system for the reactor.

The design cyclic transients for the primary system are given in FSAR Section 4.2.2. In addition, the steam generator is designed for additional conditions listed on FSAR Pages 4-11 and 4-12.

(1) The maximum steam generator differential pressure would occur with the primary system at 2750 psi and $650^{\circ}F$ and the secondary system at 770 psia and $513.8^{\circ}F$ during a turbine trip.

(2) Primary side leak testing pressure is consistent with the proposed ASME Section 11 code. The maximum temperature was that used in the fatigue analysis considering maximum NDTT shift during reactor vessel lifetime. Flooded and pressurized conditions on the secondary side assure minimum tube sheet temperature differential during leak testing.

3.1 PRIMARY COOLANT SYSTEM (Contd)

(3) The minimum temperature of $100^{\circ}F$ for pressurizing the steam generator secondary side is set by the NDTT of the manway cover of $+40^{\circ}F$.

References

(1) FSAR, Sections 6.1.2.2, 14.3.2.

(2) FSAR, Section 4.3.7.

3.1 PRIMARY COOLANT SYSTEM (Contd)

3.1.2 Heatup and Cooldown Rate

The primary coolant pressure and the system heatup and cooldown rates shall be limited in accordance with Figure 3-1, Figure 3-2 and as follows:

a. For temperatures at or below 160°F, the heatup rate shall not exceed the applicable limit line for the appropriate pressure as shown on Figure 3-1.

b. For temperatures above 160°F, the average heatup rate shall not exceed 100°F/hour.

c. Allowable combinations of pressure and temperature for a specific cooldown rate shall be below and to the right of the limit lines for that rate as shown on Figure 3-2. This rate shall not exceed 60°F/hour for temperatures at or below NDTT +60°F. The limit lines for cooling rates between those shown in Figure 3-2 may be obtained by interpolation.

d. For temperatures above NDTT +60°F, the average cooldown rate shall not exceed 100°F/hour.

e. The average heatup and cooldown rates of the pressurizer shall not exceed 200°F/hour.

f. Before the radiation exposure of the reactor vessel exceeds the exposure for which the figures apply, Figures 3-1 and 3-2 shall be updated in accordance with the following criteria and procedures:

   (1) The curve in Figure 3-3 shall be used to predict the increase in transition temperature based on integrated fast neutron flux. If measurements on the irradiation specimens show increases above this curve, a new curve shall be constructed such that it is above and to the left of all applicable data points.

   (2) Before the end of the integrated power period for which Figures 3-1 and 3-2 apply, the limit lines on the figures shall be updated for a new integrated power period as follows. The total integrated reactor thermal power from start-up to the end of the new period shall be converted to an equivalent integrated fast neutron exposure ($E \geq 1$ MeV). For this plant, 3.64×10^{19} nvt is the calculated fluence

at the reactor vessel beltline for 40 years at 2540 MWt and an 80% load factor. The predicted transition temperature increase for the end of the new period shall then be obtained from Figure 3-3, using the curve specified in 3.1.2a above.

(3) The limit lines in Figure 3-1 and Figure 3-2 shall be moved parallel to the temperature axis (horizontal) in the direction of increasing temperature a distance equivalent to the transition temperature increase during the period since the curves were last constructed. The lower vertical and intermediate vertical limit lines shall remain at $80^{\circ}F$ and $110^{\circ}F$, respectively, as they are set by the NDTT of the reactor vessel flange, steam generator and pressurizer manway covers, respectively, and are not subject to fast neutron flux. The sloping portions of the limit lines shall extend at constant slope to a temperature $140^{\circ}F$ below NDTT. At still lower temperatures, the limit lines shall be parallel to the temperature axis (horizontal) and shall intercept the sloping portions of the limit lines at NDTT $-140^{\circ}F$.

g. For plant operation until 4.9×10^5 MWD (thermal) have been accumulated, the primary system pressure during heatups and cooldowns shall be conducted in accordance with Figures 3-4 and 3-5, respectively. At that time, Figures 3-4 and 3-5 shall become void and plant operation shall be conducted in accordance with Section 3.1.2f(1).

Basis

All components in the primary coolant system are designed to withstand the effects of cyclic loads due to primary system temperature and pressure changes.[1] These cyclic loads are introduced by normal unit load transients, reactor trips and start-up and shutdown operation. During unit start-up and shutdown, the rates of temperature and pressure changes are limited. The maximum plant heatup and cooldown rate of $100^{\circ}F$ per hour is consistent with the design number of cycles and satisfies stress limits for cyclic operation.[2]

3.1 PRIMARY COOLANT SYSTEM (Contd)

The reactor vessel plate and material opposite the core has been purchased to a specified Charpy V-notch test result of 30 ft-lb or greater at an NDTT of $+10^{\circ}F$ or less, and the material has been tested to verify conformity to specified requirements and to determine the actual initial maximum NDTT value of $-30^{\circ}F$. In addition, this plate has been 100% volumetrically inspected by ultrasonic test using both longitudinal and shear wave methods. The remaining material in the reactor vessel, and other primary coolant system components, meets the appropriate design code requirements and specific component function and has a maximum NDTT of $+40^{\circ}F$.(3)

As a result of fast neutron irradiation in the region of the core, there will be an increase in the NDTT with operation. The techniques used to predict the integrated fast neutron ($E > 1$ MeV) fluxes of the reactor vessel are described in Section 3.3.2.6 of the FSAR and also in Amendment 13, Section II.

Since the neutron spectra and the flux measured at the samples and reactor vessel inside radius should be nearly identical, the measured transition shift for a sample can be applied to the adjacent section of the reactor vessel for later stages in plant life equivalent to the difference in calculated flux magnitude. The maximum exposure of the reactor vessel will be obtained from the measured sample exposure by application of the calculated azimuthal neutron flux variation. The maximum integrated fast neutron ($E > 1$ MeV) exposure of the reactor vessel is computed to be 3.64×10^{19} nvt^{2} for 40 years' operation at 2540 MWt and 80% load factor.(7) The predicted NDTT shift for an integrated fast neutron ($E > 1$ MeV) exposure of 3.64×10^{19} nvt^{2} is $260^{\circ}F$, the value obtained from the curve shown in Figure 3-3.(3) The actual shift in NDTT will be established periodically during plant operation by testing of reactor vessel material samples which are irradiated cumulatively by securing them near the inside wall of the reactor vessel as described in Section 4.5.3 and Figure 4-11 of the FSAR. To compensate for any increase in the NDTT caused by irradiation, limits on the pressure-temperature relationship are periodically changed to stay within the stress limits during heatup and cooldown. During the

3.1 PRIMARY COOLANT SYSTEM (Contd)

first two years of reactor operation, a conservatively high fluence of 1.82 x 10^{18} nvt is assumed which corresponds to 2540 MWt for 584 days. The corresponding NDTT shift is 80°F, based on the curve shown in Figure 3-3. Thus, for this interval, the upper limit to the NDTT is (initial + shift) or +10 + 80 = 90°F. The stress allowed in the reactor vessel(1) in relation to NDTT to preclude the possibility of brittle failure is:

a. Above NDTT +60°F, no limitation because of brittle fracture effects.

b. At NDTT +60°F, use 20% of yield stress Sy obtained from Figure 12 of Reference (8).

c. At NDTT, use 16% of yield stress Sy obtained from Figure 12 of Reference (8).

d. At NDTT -140°F and below, use 10% of yield stress Sy obtained from Figure 12 of Reference (8).

These stress limitations are established by assuming a maximum flaw of the size used in the reference specimen of the nondestructive test method. For this reactor vessel, which will have a nominal shell thickness at the heaviest section of about 10-3/4 inches, a strength reduction factor of 20%, as indicated above, will be used to account for possible effects of the maximum size flaw which might not be detected by the nondestructive examination. The 10% factor used in the lower temperature ranges provides additional conservatism. The allowable yield stress shall have a linear variation between Points b, c and d above.

The limit lines in Figures 3-1 and 3-2 are based on these stress limits and contain allowances for a 10°F margin between actual and measured temperature and 50 psi margin between actual and measured pressure. During cooldown, the thermal stress varies from tensile at the inner wall to compressive at the outer wall. The internal pressure superimposes a tensile stress on this thermal stress pattern, increasing the stress at the inside wall and relieving the stress at the outside wall. Therefore, for this range of cooldown rates, the limiting stress always appears at the inside wall so the limit line has a direct dependence on cooldown rate. This leads to a family of curves for cooldown as shown on Figure 3-2.

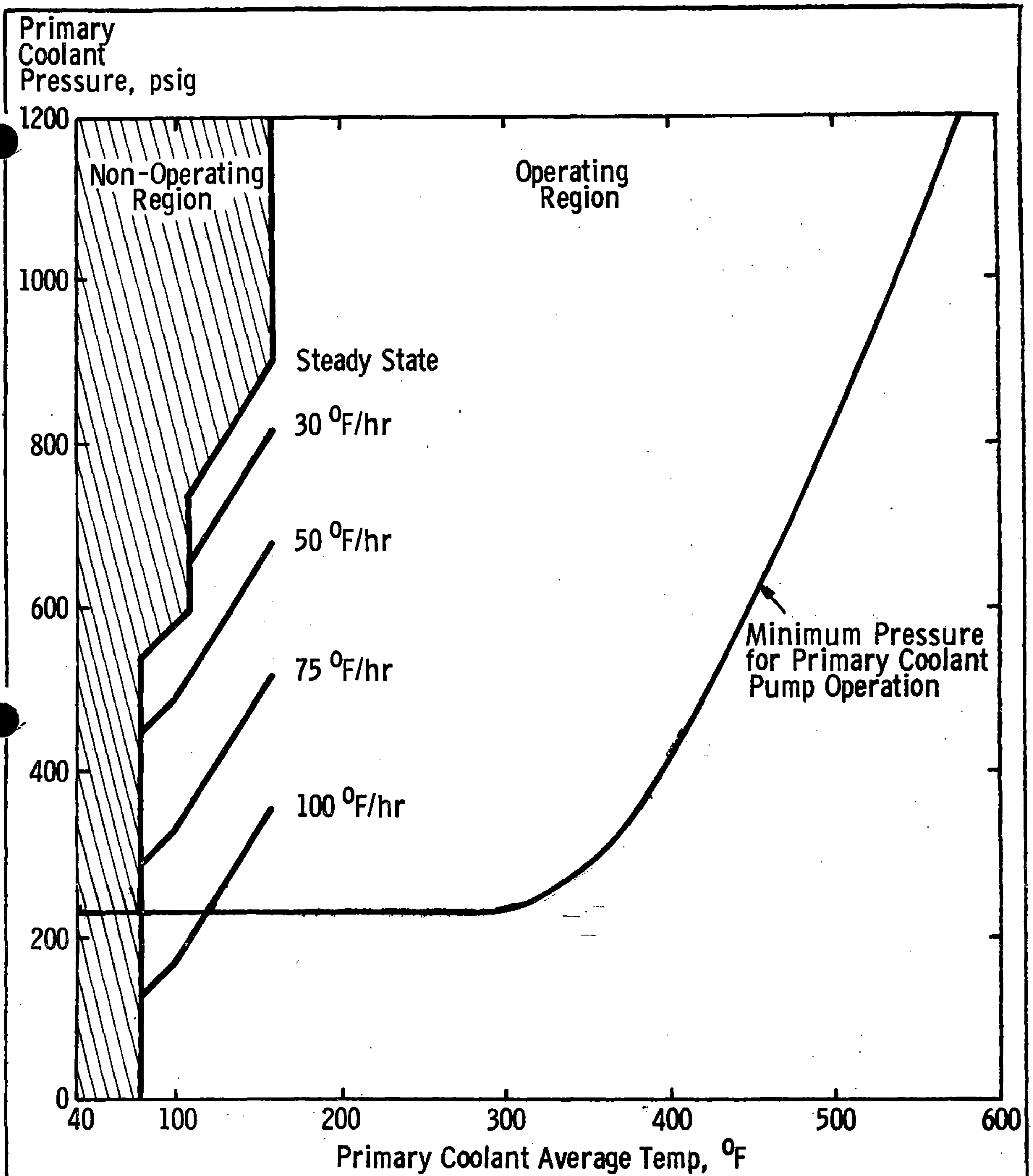
3.1 PRIMARY COOLANT SYSTEM (Contd)

For heatup, the thermal stresses are reversed and the location of the limiting stress is a function of the heatup rate. The limit lines for heatup rates are shown on Figure 3-1.

Figures 3-1 and 3-2 define stress limitations only. For normal operation other inherent plant characteristics, eg, pump parameters, pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure ranges. The normal maximum heatup and cooldown rate for the primary coolant system is approximately $60^{\circ}F$ per hour.

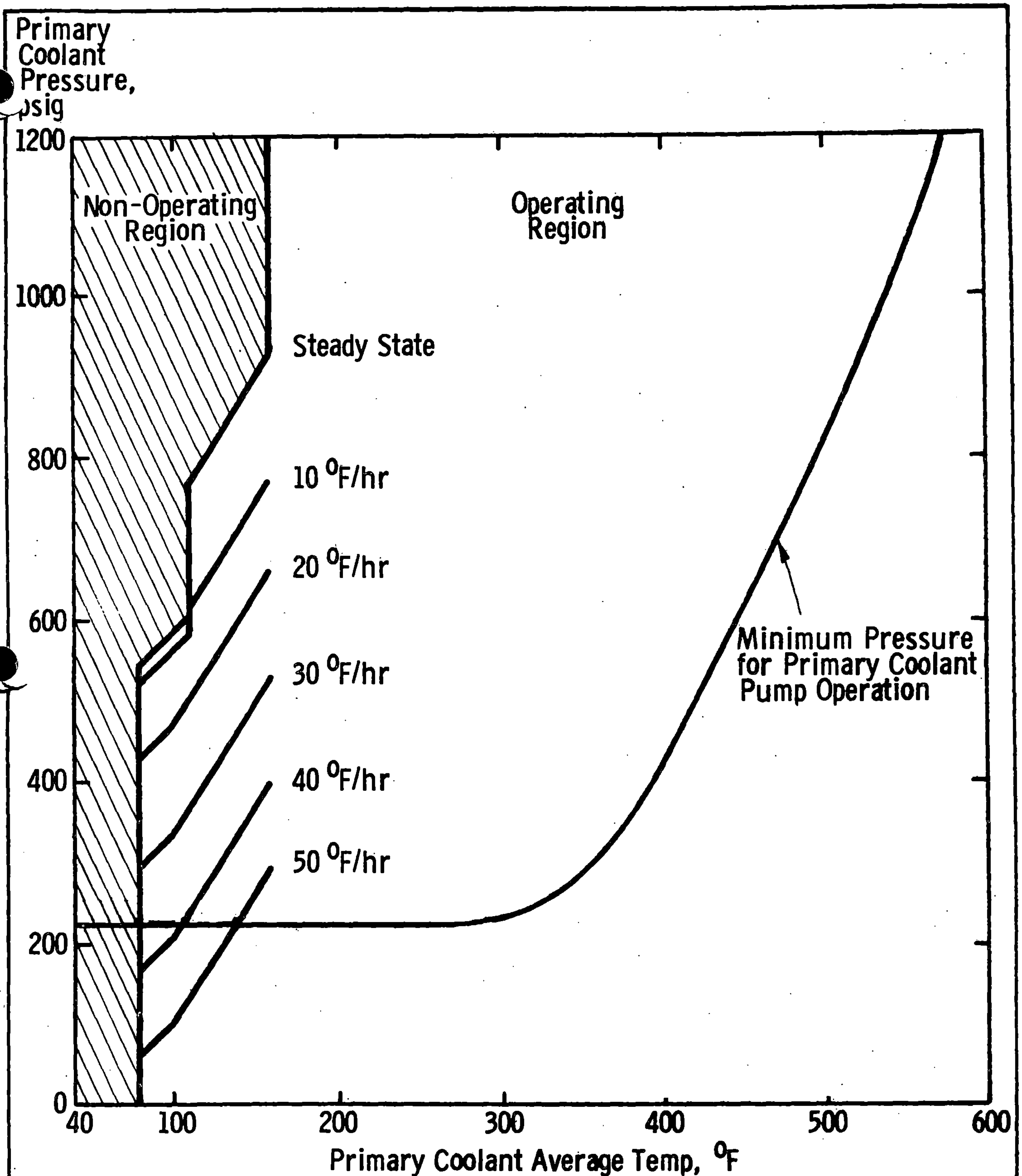
As was stated above in these bases, the reactor vessel plate material in the core region has an initial NDTT of $+10^{\circ}F$ and possibly as low as a $-30^{\circ}F$. In order to take advantage of this property during the early phases of plant operation when heatups and cooldowns and pressurizations will be more frequent, the amount of operation to give a $30^{\circ}F$ shift in NDTT (to $+40^{\circ}F$) has been calculated. This value is 6×10^{17} nvt fluence which corresponds to 4.9×10^{5} MWD (thermal) using the design shift curve given on Figure 3-3 of this specification. This shift in NDTT is considered to be a conservative estimate.[9,10]

Assuming the reactor vessel initial NDTT is $+10^{\circ}F$, the $+30^{\circ}F$ shift will take the reactor vessel to $+40^{\circ}F$, which is the same as the maximum NDTT value of the other components in the primary coolant system. These other components establish a minimum pressurization temperature (NDTT $+60^{\circ}F$) of $100^{\circ}F$ ($40^{\circ}F + 60^{\circ}F$); thus, at 4.9×10^{5} MWD (thermal), the reactor vessel becomes controlling for minimum pressurization temperature. A shift to the 2-year curves, Figures 3-1 and 3-2, will be made at this time.



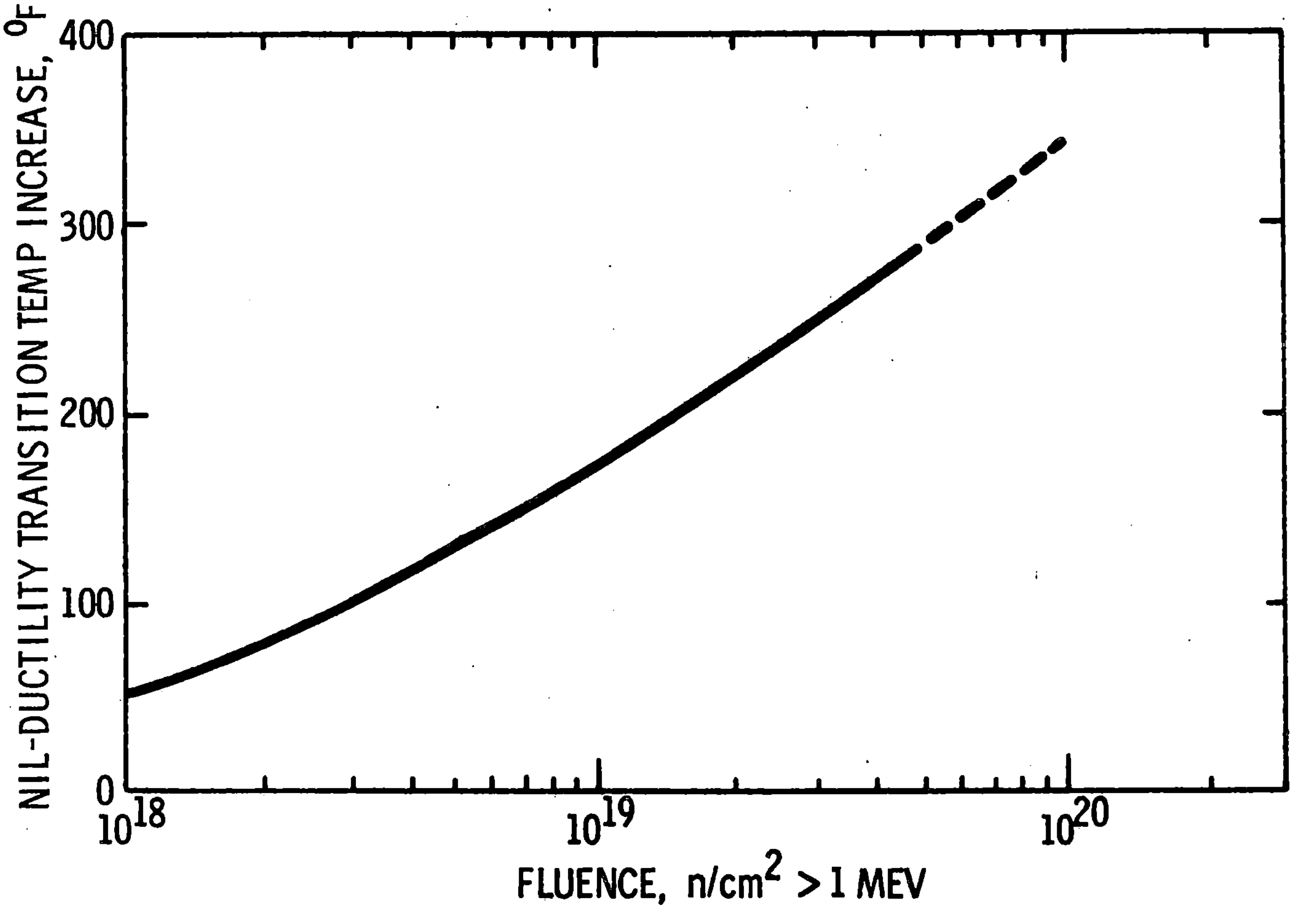
Pressure - Temperature Relationship During Plant Heatup	Palisades Technical Specifications	Figure 3-1

Rev 1/70

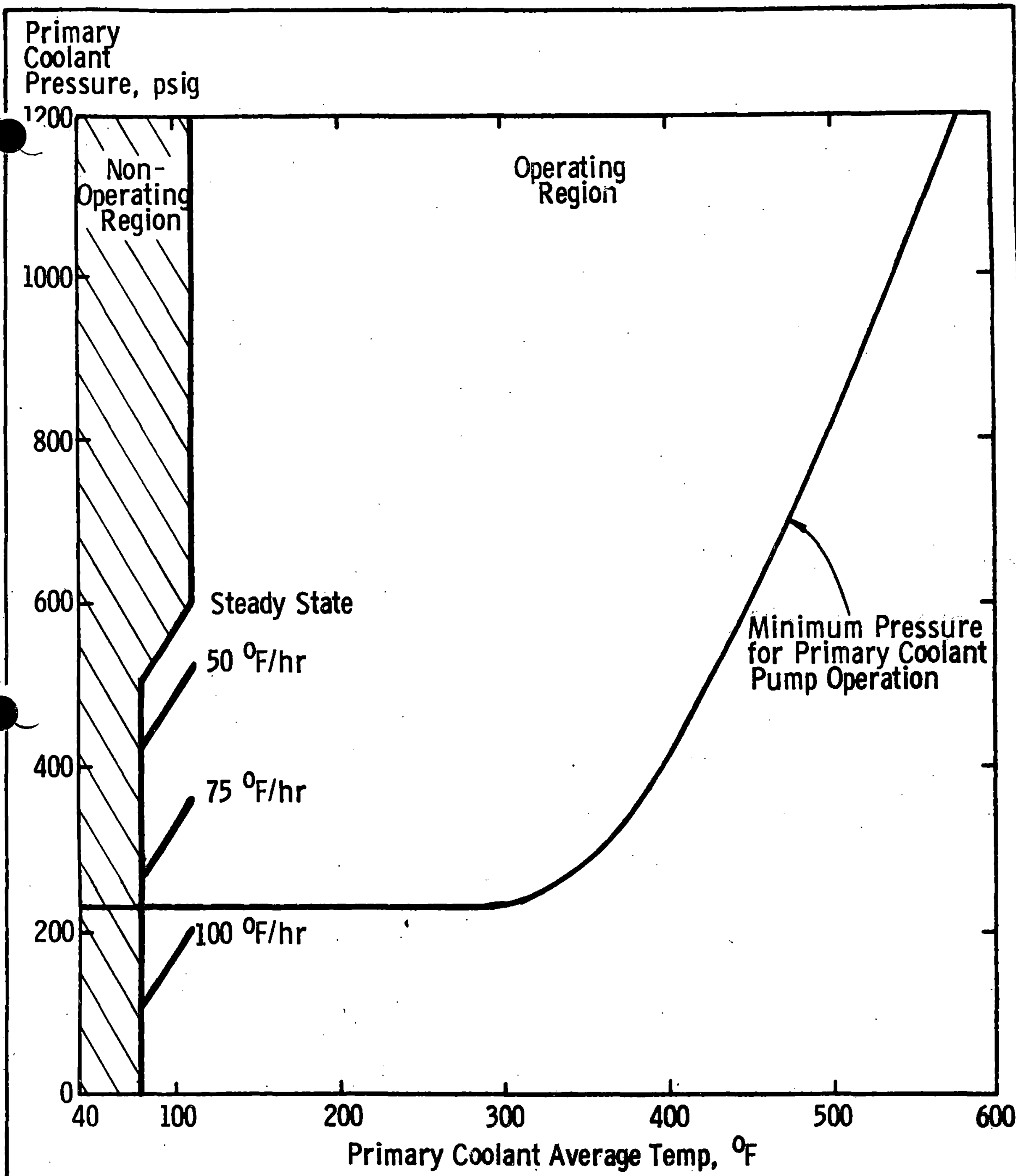


Pressure - Temperature Relationship During Plant Cooldown	Palisades Technical Specifications	Figure 3-2

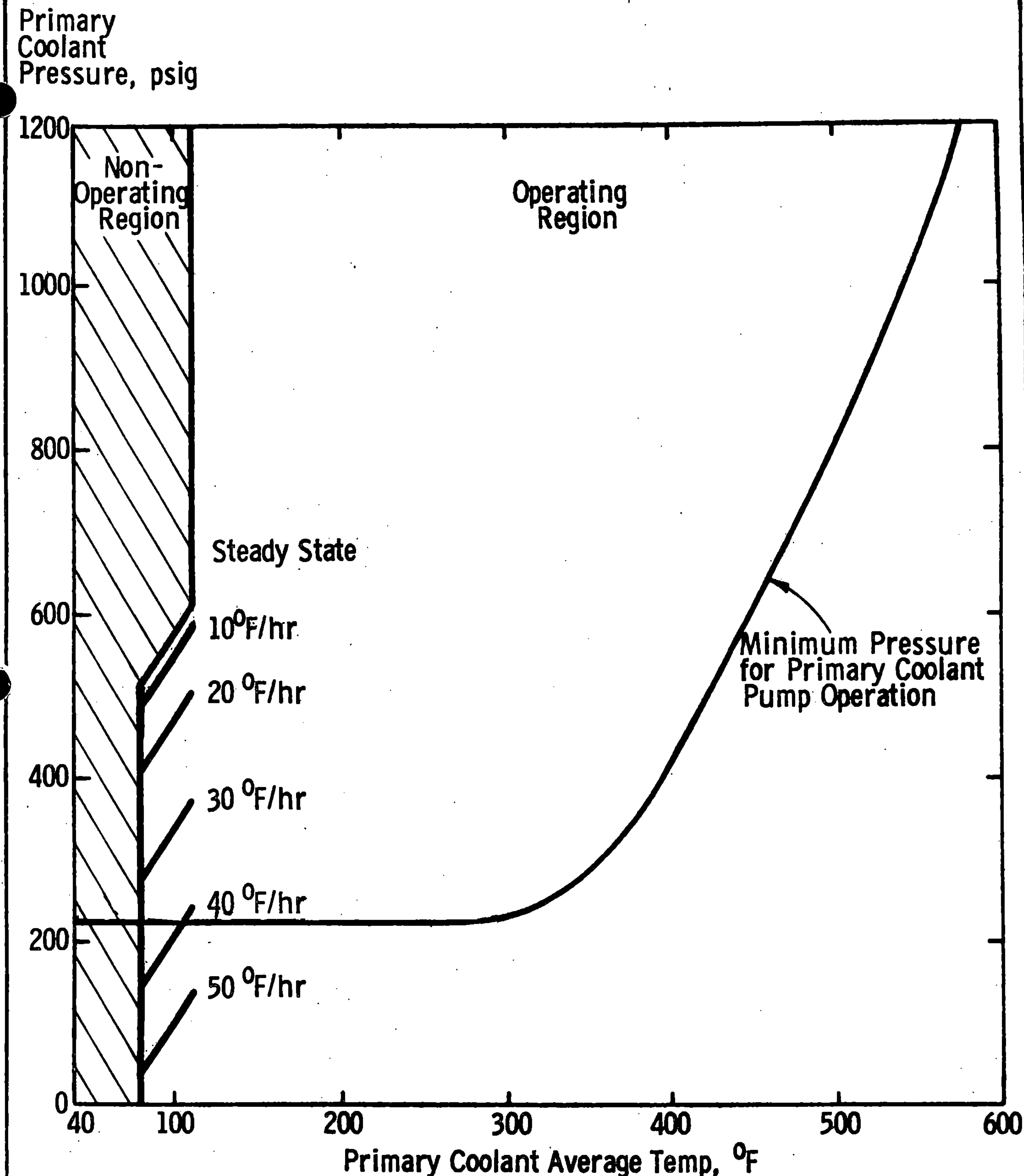
Rev 1/70



Design Curve of NDTT Increase (550° F Irradiation)	Palisades Technical Specifications	Figure 3-3



Pressure - Temperature Relationship During Plant Heatup (For First 4.9 x 10^5 MWD(t))	Palisades Technical Specifications	Figure 3-4



Pressure - Temperature Relationship During Plant Cooldown (For First 4.9×10^5 MWD(t))	Palisades Technical Specifications	Figure 3-5

3.1 PRIMARY COOLANT SYSTEM (Contd)

References

(1) FSAR, Section 4.2.2.

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

(3) FSAR, Section 4.2.4.

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

(5) FSAR, Section 4.3.1.

(6) FSAR, Section 4.4.1.

(7) FSAR, Amendment 15.

(8) Technical Paper - WAPD-BT18 Bettis Technical Review, Reactor Technology Section, April 1960 - Application of Stress Concentration Factors by B. F. Langer.

(9) NRL Report No 6772.

(10) Figure 4.2.3, PSAR - Nuclear No 4 & 5, Docket No 50-342 & 50-343, Consolidated Edison Company of New York.

3.1 PRIMARY COOLANT SYSTEM (Contd)

3.1.3 Minimum Conditions for Criticality

a. Except during low-power physics tests, the reactor shall not be made critical if the primary coolant temperature is below 525°F.

b. In no case shall the reactor be made critical if the primary coolant temperature is below NDTT +120°F.

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

d. No more than one control rod at a time shall be exercised or withdrawn until after a steam bubble and normal water level are established in the pressurizer.

e. Primary coolant boron concentration shall not be reduced until after a steam bubble and normal water level are established in the pressurizer.

Basis

At the beginning of life of the initial fuel cycle, the moderator temperature coefficient is expected to be slightly negative at operating temperatures with all control rods withdrawn.(1) However, the uncertainty of the calculation is such that it is possible that a slightly positive coefficient could exist.

The moderator coefficient at lower temperatures will be less negative or more positive than at operating temperature.(1, 2) It is therefore prudent to restrict the operation of the reactor when primary coolant temperatures are less than normal operating temperature ($\geq$525°F).

Assuming the most pessimistic rods out moderator coefficient, the maximum potential reactivity insertion that could result from depressurizing the coolant from 525°F to saturation temperature at 2100 psia is 0.1% $\Delta\rho$.

During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient(3) and the small integrated $\Delta\rho$ would limit the magnitude of a power excursion resulting from a reduction of moderator density.

3.1 PRIMARY COOLANT SYSTEM (Contd)

The requirement that the reactor is not to be made critical below NDTT +120°F provides increased assurance that the proper relationship between primary coolant pressure and temperature will be maintained relative to the NDTT of the primary coolant system. Heatup to this temperature will be accomplished by operating the primary coolant pumps.

If the shutdown margin required by Specification 3.10.1 is maintained, there is no possibility of an accidental criticality as a result of an increase of moderator temperature or a decrease of coolant pressure. Normal water level is established in the pressurizer prior to the withdrawal of control rods or the dilution of boron so as to preclude the possible overpressurization of a solid primary coolant system.

References

(1) FSAR, Table 3-2.

(2) FSAR, Table 3-6.

(3) FSAR, Table 3-3.

(4) FSAR Section 4.3.7

3.1 PRIMARY COOLANT SYSTEM (Contd)

3.1.4 Maximum Primary Coolant Radioactivity

Specifications

The total specific radioactivity of the primary coolant due to nuclides with half-lives of more than 30 minutes shall not exceed $75/\bar{E}$ μCi/cc-MeV at 568°F whenever the average primary coolant temperature is greater than 525°F (where $\bar{E}$ is the measured average of the beta and gamma energies per disintegration in MeV).

Basis

The maximum specified primary coolant radioactivity is intended to be that which would result from operation with 0.85% fuel defects.(1) Radiation shielding and the radioactive waste disposal systems were designed to operate with 1% defects(2) and, therefore, with the specified maximum radioactivity, these considerations are not limiting. The specified limit provides protection to the public against the potential release of primary coolant radioactivity to the atmosphere as demonstrated by the following analysis of a steam generator tube rupture accident.(3)

The maximum potential dose at the site boundary due to this accident, based upon an upper limit calculation, is larger and hence more limiting than the dose that would result from one year of operation with the maximum unidentified leakage from the primary coolant system allowable by Specification 3.1.5a.

Rupture of a steam generator tube would cause a reactor and turbine trip and allow primary coolant radioactivity to enter the secondary system. Calculations indicate that about 71% of the total $\bar{E}$ is due to radioactive noble gases.(1) These would be released to the atmosphere from the condenser air ejector and relief of atmospheric dump valves. For purposes of the accident dose calculation, however, it is conservatively assumed that 100% of the radioactivity is released from that portion of the primary coolant leaking into the secondary system. Radioactivity would be released until the operator could reduce the primary coolant system pressure below the set point of the secondary system relief valves and could isolate the faulty steam generator. The accident is considered to be a double-ended break of a single steam generator tube followed by initiation of system cooldown within

3.1 PRIMARY COOLANT SYSTEM (Contd)

30 minutes after the tube break and complete isolation of both steam generators within 3.6 hours. During this period, approximately one-third of the total primary coolant and its associated radioactivity is released to the secondary system.(3)

The whole-body dose resulting from immersion in the accident released cloud containing the released radioactivity would include both gamma and beta radiation. The gamma dose is dependent on the finite size and configuration of the cloud. However, the analysis employs the simple model of a semi-infinite cloud, which gives an upper limit to the potential gamma dose. The semi-infinite cloud model is applicable to the beta dose because of the short range of beta radiation in air. It is further assumed that meteorological conditions during the course of the accident correspond to Pasquill Type F and 2 meters per second wind speed, resulting in an X/Q value of 2.64×10^{-4} sec/m^3. This includes an appropriate building wake effect associated with a cross-sectional area of 2210 m^2.

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

$$\text{Dose (Rem)} = 1/2 \left[\bar{\bar{E}} \cdot A \cdot V/3 \cdot X/Q \cdot (3.7 \times 10^{10}) \cdot (1.33 \times 10^{-11})\right]$$

Where: $\bar{\bar{E}}$ = average energy of betas and gammas per disintegration (MeV)

A = primary coolant activity ($75/\bar{\bar{E}}$ Ci/m^3-MeV)

V = primary coolant volume (310 m^3); V/3 is volume of coolant released

X/Q = 2.6×10^{-4} sec/m^3, 0-2 hr dispersion coefficient of the site boundary (677 m) using an appropriate building wake coefficient.

3.7×10^{10} dps/Ci

1.33×10^{-11} Rem/MeV/m^3

The resulting whole-body dose is 0.5 Rem.

3.1 PRIMARY COOLANT SYSTEM (Contd)

The 525°F temperature in the specification corresponds to a saturation pressure of 848 psia, which is below the 985 psia minimum set point of the secondary system relief valves. Therefore, potential primary to secondary leakage at temperatures below 525° could be contained within the steam generator by closing the steam line isolation valve on the defective steam generator.

The 568°F temperature in the specification corresponds to the average temperature of the primary coolant at rated operating conditions. Therefore, measurements of primary coolant radioactivity concentrations made at room temperature will be density corrected to 568°F.

Measurement of $\bar{E}$ will be performed at least twice annually, and in any event will be performed each time the primary coolant radioactivity concentration changes by 10 μCi/cc from the previous measurement. Calculations required to determine $\bar{E}$ will consist of the following:

1. Quantitative measurement in units of μCi/cc of radionuclides with half-lives longer than 30 minutes making up at least 95% of the total activity in the primary coolant.
2. A determination of the beta and gamma decay energy per disintegration of each nuclide determined in (1) above by applying known decay energies and schemes. (Table of Isotopes, Sixth Edition, March 1968.)
3. A calculation of $\bar{E}$ by appropriate weighting of each nuclide's beta and gamma energy with its concentration as determined in (1) above.

References

(1) FSAR, Table II-1.
(2) FSAR, Section II.1.1.
(3) FSAR, Section 14.15.

3.1 PRIMARY COOLANT SYSTEM (Contd)

3.1.5 Primary Coolant System Leakage Limits

Specifications

a. If the primary coolant system leakage exceeds 1 gpm and the source of leakage is not identified, the reactor shall be placed in the hot shutdown condition within 12 hours and cold shutdown condition within 24 hours.

b. If leakage from the primary coolant system exceeds 10 gpm, the reactor shall be placed in the hot shutdown condition within 12 hours and cold shutdown condition within 24 hours.

c. If the radioactivity of the secondary coolant in a steam generator exceeds 0.5 μCi of I-131/cm^3 at 518^{o}F, the reactor shall be placed in the hot shutdown condition within 12 hours.

Basis

Leakage directly into the containment indicates the possibility of a breach in the primary coolant envelope. The limitation of 1 gpm for a source of leakage not identified is sufficiently above the minimum detectable leakage rate to provide a reliable indication of leakage.(1) The limit is held low to minimize the chance of a crack progressing to an unsafe condition without detection and proper evaluation by the technical staff.

When the source of leakage can be identified, the situation shall be evaluated to determine if operation can safely continue. This evaluation will be performed by the Plant Operating Staff and will be documented in writing and approved by either the Plant Superintendent or Assistant Plant Superintendent. Under these conditions, a maximum allowable primary coolant leakage rate of 10 gpm has been established. This does not include the primary coolant pump seal leak off that is piped to the volume control tank, which is not considered "leakage" from the primary coolant system. A primary coolant leakage to the containment atmosphere greater than 10 gpm would be indicative of seal and packing failures of sufficient magnitude to warrant shutdown for repair.

The maximum primary coolant leakage rate of 10 gpm is within the 40 gpm capacity of one charging pump which would be available even under a

3.1 PRIMARY COOLANT SYSTEM (Contd)

loss-of-off-site power condition. Any primary coolant leakage to the containment atmosphere may be identified by the containment building gas monitor, which is sensitive to low leak rates and can sample any of the containment cooler fan discharges, or by containment humidity detectors. The sensitivity of the instrument is expected to be less than 0.1 gpm based on detection of short-lived activation gases.[1, 2] If primary coolant leakage is to another closed system, it can be detected by the plant radiation monitors or by inventory control. Placing the reactor in hot shutdown within 12 hours provides adequate time to arrange for an orderly reduction of power on the plant. The hot shutdown condition allows personnel to enter the containment and to inspect the pressure boundary for leaks. The 24 hours allowed prior to reaching a cold shutdown condition allows reasonable time to correct small deficiencies. If major repairs are needed, a cold shutdown condition would be in order.

The limiting secondary coolant radioactivity concentration is based upon potential off-site doses calculated for a complete loss-of-load incident, using the following conservative assumptions:

1. A total of 125 m^3 of secondary coolant is released (approximate volume of coolant in both steam generators) although accident analyses indicate that only 43.5 m^3 would be released.
2. One-tenth of the coolant iodine inventory reaches the site boundary, which assumes minimum credit for steam generator moisture separator effectiveness and other iodine removal mechanisms.
3. Pasquill Type F meteorology.

The limiting dose for this case would be due to the iodine because of its low MPC in air, for which the inhalation dose at the site boundary is computed as follows:

$$\text{Dose (Rem)} = C \cdot V \cdot B(t) \cdot X/Q \cdot DCF \cdot 1.5 \cdot 10^{-1}$$

Where: C = secondary coolant I-131 activity (0.5 Ci/m^3)

V = secondary coolant volume released (125 m^3)

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

X/Q = 2.6×10^{-4} sec/m^3, 0-2 hr dispersion coefficient at site boundary (677 m) using a building wake coefficient of 1/2

3.1 PRIMARY COOLANT SYSTEM (Contd)

DCF = 1.48×10^6 Rem/Ci I-131 inhaled

1.5 = the contribution factor of I-133 to the inhalation dose

10^{-1} = fraction of iodine airborne to site boundary

The resultant thyroid dose is less than 1.5 Rem.

The 518°F in the specification corresponds to the average temperature of the secondary coolant at rated operating conditions. Therefore, measurements of radioactivity concentrations made at room temperature will be density corrected to 518°F.

References

(1) FSAR, Amendment 15, Question 4.3.

(2) FSAR, Section 11, Table 11-6.

3.1 PRIMARY COOLANT SYSTEM (Contd)

3.1.6 Maximum Primary Coolant Oxygen and Halogens Concentrations

Specifications

a. If the concentration of oxygen in the primary coolant exceeds 0.1 ppm during power operation, corrective action shall be initiated within eight hours to return oxygen levels to $\leq$ 0.1 ppm.

b. If the concentration of chloride in the primary coolant exceeds 0.12 ppm during power operation, corrective action shall be initiated within eight hours to return chloride levels to $\leq$ 0.12 ppm.

c. If the concentration of fluorides in the primary coolant exceeds 0.10 ppm following modifications or repair to the primary system involving welding, corrective action shall be initiated within eight hours to return fluoride levels to $\leq$ 0.10 ppm.

d. If the oxygen concentration and the chloride or fluoride concentration of the primary coolant simultaneously exceed the limits given in (a), (b) and (c), respectively, corrective action is to be taken immediately to return the system to within normal operation specifications.

e. If the concentration limits of oxygen and chloride or fluoride given in (a), (b) and (c) above are not restored within 24 hours, the reactor shall be placed in a hot shutdown condition within 12 hours thereafter. If the normal operational limits are not restored within an additional 24-hour period, the reactor shall be placed in a cold shutdown condition within 24 hours thereafter.

Basis

By maintaining the oxygen, chloride and fluoride concentration in the primary coolant within the limits specified, the integrity of the primary coolant system is protected against potential stress corrosion attack.[1,2]

If these limits are exceeded, measures can be taken to correct the condition (eg, replacement of ion exchange resin or adjustment of the hydrogen concentration in the volume control tank) and further, because of the time-dependent nature of any adverse effects arising from the oxygen or halogens concentration in excess of the limits, it is unnecessary to shut down immediately since the condition can be corrected.

3.1 PRIMARY COOLANT SYSTEM (Contd)

The oxygen and halogen limits specified are at least an order of magnitude below concentration which could result in damage to the materials found in the primary coolant system even if maintained for an extended period of time.[3] Thus, the period of eight hours to initiate corrective action to restore individual concentrations within the limits, or 24 hours to restore both concentrations has been established. If the corrective action has not been effective at the end of the 24-hour period, then the reactor will be brought to the hot shutdown condition and the corrective action will continue. If, at the end of a further 24-hour period, the corrective action has not been effective, long-term corrective action could be required and the reactor will be brought to the cold shutdown condition.

References

(1) FSAR, Section 4.3.11, 4.3.13.

(2) FSAR, Section 9.10.

(3) Corrosion & Wear Handbook, O. J. DePaul, Editor

3.1 PRIMARY COOLANT SYSTEM (Contd)

3.1.7 Primary and Secondary Safety Valves

Specifications

a. The reactor shall not be made critical unless all three pressurizer safety valves are operable with their lift settings maintained between 2500 psia and 2580 psia.

b. A minimum of one operable safety valve shall be installed on the pressurizer whenever the reactor head is on the vessel.

c. Whenever the reactor is in power operation, a minimum of 22 secondary system safety valves shall be operable with their lift settings between 985 psia and 1025 psia.

Basis

The primary and secondary safety valves pass sufficient steam to limit the primary system pressure to 110 percent of design (2750 psia) following a complete loss of turbine generator load without simultaneous reactor trip while operating at 2650 MWt.[1] The reactor is assumed to trip on a "High Primary Coolant System Pressure" signal. To determine the maximum steam flow, the only other pressure relieving system assumed operational is the secondary system safety valves. Conservative values for all system parameters, delay times and core moderator coefficient are assumed. Overpressure protection is provided to the portions of the primary coolant system which are at the highest pressure considering pump head, flow pressure drops and elevation heads.

If no residual heat were removed by any of the means available, the amount of steam which could be generated at safety valve lift pressure would be less than half of one valve's capacity. One valve, therefore, provides adequate defense against overpressurization when the reactor is subcritical.

The total relief capacity of the 24 secondary system safety valves is 11.7×10^6 lb/hr. This is based on a steam flow equivalent to an NSSS power level of 2650 MWt at the nominal 1000 psia valve lift pressure. At the initial rated power of 2200 MWt, a relief capacity of only 9.8×10^6 lb/hr is required to prevent overpressurization of the secondary system on loss-of-load conditions and 22 valves provide relieving capability of 10,705,200 lb/hr.[1]

Reference

(1) FSAR, Sections 4.3.4, 4.3.7 and 14.12.4.

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 assure safe reactor operation.

Specifications

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

3.2.2 The reactor shall not be made critical unless all the following conditions are met:

a. At least two charging pumps shall be operable.

b. One concentrated boric acid transfer pump shall be operable.

c. The two concentrated boric acid tanks together shall contain a minimum of 118 inches of a 6-1/4 percent to 10 percent by weight boric acid solution at a temperature of at least 25°F above saturation temperature for the concentration present in the tank.

d. System piping and valves shall be operable to the extent of establishing two flow paths from the concentrated boric acid tanks to the primary coolant system and a flow path from the SIRW tank to the charging pumps.

e. Both channels of heat tracing shall be operable for the above flow paths.

3.2.3 During power operation, the requirements of 3.2.2 may be modified to allow any one of the following conditions to be true at any one time. If the system is not restored to meet the requirements of 3.2.2 within the time period specified, the reactor shall be placed in a hot shutdown condition within 12 hours. If the requirements of 3.2.2 are not satisfied within an additional 48 hours, the reactor shall be placed in a cold shutdown condition within 24 hours.

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

b. One concentrated boric acid tank may be out of service provided a minimum of 118 inches of 6-1/4% to 10% by weight boric acid

3.2 CHEMICAL AND VOLUME CONTROL SYSTEM (Contd)

solution at a temperature of at least 25°F above saturation temperature is contained in the operable tank and provided that the tank is restored to operable status within 24 hours.

c. Only one flow path from the concentrated boric acid tanks to the primary coolant system may be operable provided that either the other flow path from the concentrated boric acid tanks to the primary coolant system or flow path from the SIRW tank to the charging pumps is restored to operable status within 24 hours.

d. One channel of heat tracing may be out of service provided it is restored to operable status within 24 hours.

Basis

The chemical and volume control system provides control of the primary coolant system boron inventory.(1) This is normally accomplished by using any one of the three charging pumps in series with one of the two boric acid pumps. An alternate method of boration will be to use the charging pumps directly from the SIRW storage tank. A third method will be to depressurize and use the safety injection pumps. There are two sources of borated water available for injection through three different paths.

a. The boric acid transfer pumps can deliver the concentrated boric acid tank contents (6-1/4 - 10 percent concentration of boric acid) to the charging pumps.

b. The safety injection pumps can take suction from the SIRW tank (1720 ppm boron solution).

c. The charging pumps can take their suctions by gravity from either the boric acid tanks or the SIRW tank.

Each concentrated boric acid tank containing 118 inches of 6-1/4 weight percent boric acid has sufficient boron to bring the plant to a cold shutdown condition. Boric acid pumps are each of sufficient capacity to feed all three charging pumps at their maximum capacity.

The concentrated boric acid storage tank is sized for 6-1/4 weight percent boric acid solution and is capable of storing solution up to 12 weight percent. All components of the system are capable of maintaining 12 weight percent solution.

3.2 CHEMICAL AND VOLUME CONTROL SYSTEM (Contd)

Duplicate heating equipment is provided on all components of the system to maintain the surface temperature to at least 150°F, which is 30°F above the saturation temperature of a 10% solution. If the heater elements fail to maintain 150°F, sufficient time is available to energize the redundant heater elements before the 25°F limit above saturation temperature is reached. Also, the 25°F limit provides assurance that the plant can be shut down before the precipitation temperature is reached.

The SIRW tank contents are sufficient to borate the primary coolant in order to reach cold shutdown at any time during core life.

The limits on which components may be inoperable and the time periods for inoperability were selected on the basis of the redundancy indicated above and engineering judgment.

Reference

(1) FSAR, Section 9.10.

3.3 EMERGENCY CORE COOLING SYSTEM

Applicability

Applies to the operating status of the emergency core cooling system.

Objective

To assure operability of equipment required to remove decay heat from the core in either emergency or normal shutdown situations.

Specifications

Safety Injection and Shutdown Cooling Systems

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

a. The SIRW tank contains not less than 250,000 gallons of water with a boron concentration of at least 1720 ppm at a temperature not less than $40^{\circ}F$.

b. All four safety injection tanks are operable and pressurized to at least 200 psig with a tank liquid level of at least 187 inches and a maximum level of 196 inches with a boron concentration of at least 1720 ppm.

c. One low-pressure safety injection pump is operable on each bus.

d. One high-pressure safety injection pump is operable on each bus.

e. Both shutdown heat exchangers and both component cooling heat exchangers are operable.

f. Piping and valves shall be operable to provide two flow paths from the SIRW tank to the primary coolant system.

g. All valves, piping and interlocks associated with the above components and required to function during accident conditions are operable.

3.3.2 During power operation, the requirements of 3.3.1 may be modified to allow one of the following conditions to be true at any one time. If the system is not restored to meet the requirements of 3.3.1 within the time period specified below, the reactor shall be placed in a hot shutdown condition within 12 hours. If the requirements of 3.3.1 are not met within an additional 48 hours, the reactor shall be placed in a cold shutdown condition within 24 hours.

a. One safety injection tank may be inoperable for a period of no more than one week.

b. One low-pressure safety injection pump may be inoperable provided the pump is restored to operable status within 24 hours.

3.3 EMERGENCY CORE COOLING SYSTEM (Contd)

The other low-pressure safety injection pump shall be tested to demonstrate operability prior to initiating repair of the inoperable pump.

c. One high-pressure safety injection pump may be inoperable provided the pump is restored to operable status within 24 hours. The other high-pressure safety injection pump shall be tested to demonstrate operability prior to initiating repair of the inoperable pump.

d. One shutdown heat exchanger and one component cooling water heat exchanger may be inoperable for a period of no more than 24 hours.

e. Any valves, interlocks or piping directly associated with one of the above components and required to function during accident conditions shall be deemed to be part of that component and shall meet the same requirements as listed for that component.

f. Any valve, interlock or pipe associated with the safety injection and shutdown cooling system and which is not covered under 3.3.2e above but, which is required to function during accident conditions, may be inoperable for a period of no more than 24 hours. Prior to initiating repairs, all valves and interlocks in the system that provide the duplicate function shall be tested to demonstrate operability.

Basis

The normal procedure for starting the reactor is, first, to heat the primary coolant to near operating temperature by running the primary coolant pumps. The reactor is then made critical by withdrawing control rods and diluting boron in the primary coolant.[1] With this mode of start-up, the energy stored in the primary coolant during the approach to criticality is substantially equal to that during power operation and, therefore, all engineered safety features and auxiliary cooling systems are required to be fully operable. During low-temperature physics tests, there is a negligible amount of stored energy in the primary coolant; therefore, an accident comparable in

3.3 EMERGENCY CORE COOLING SYSTEM (Contd)

severity to the design basis accident is not possible and the engineered safeguards' systems are not required.

The SIRW tank contains a minimum of 250,000 gallons of water containing 1720 ppm boron. This is sufficient boron concentration to provide a 5% shutdown margin with all control rods withdrawn and a new core at a temperature of $60^{O}F$.

Heating steam is provided to maintain the tank above $40^{O}F$ to prevent freezing. The 1% boron (1720 ppm) solution will not precipitate out above $32^{O}F$. The source of steam during normal plant operation is extraction steam line in the turbine cycle.

The limits for the safety injection tank pressure and volume assure the required amount of water injection during an accident and are based on values used for the accident analyses. The minimum 187-inch level corresponds to a volume of 1067 ft^3 and the maximum 196-inch level corresponds to a volume of 1087 ft^3.

Prior to the time the reactor is brought critical, the valving of the safety injection system must be checked for correct alignment and appropriate valves locked. Since the system is used for shutdown cooling, the valving will be changed and must be properly aligned prior to start-up of the reactor.

The operable status of the various systems and components is to be demonstrated by periodic tests. 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, 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) is to be tested prior to initiating repair of the inoperable component. If it develops that (a) the inoperable component is not repaired within the specified allowable time period; or (b) a second component in the same or related system is found to be inoperable, the reactor will initially be put in the hot shutdown

3.3 EMERGENCY CORE COOLING SYSTEM (Contd)

condition to provide for reduction of the decay heat from the fuel and consequent reduction of cooling requirements after a postulated loss-of-coolant accident. This will also permit improved access for repairs in some cases. After a limited time in hot shutdown, if the malfunction(s) is not corrected, the reactor will be placed in the cold shutdown condition utilizing normal shutdown and cooldown procedures. In the cold shutdown condition, release of fission products or damage of the fuel elements is not considered possible.

The plant operating procedures will require immediate action to effect repairs of an inoperable component and, therefore, in most cases, repairs will be completed in less than the specified allowable repair times. The limiting times to repair are intended to: (1) Assure that operability of the component will be restored promptly and yet, (2) allow sufficient time to effect repairs using safe and proper procedures.

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 reduces the 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.

With respect to the core cooling function, there is functional redundancy over most of the range of break sizes.[2]

Adequate core cooling for the break spectrum up to and including the 42-inch double-ended break is assured with the minimum safety injection which is defined as follows: For the system of four passive safety injection tanks, the entire contents of one tank are assumed to be unavailable for emergency core cooling. In addition, of the three high-pressure safety injection pumps and the two low-pressure safety injection pumps, only one of each type is assumed to operate; and, also

3.3 EMERGENCY CORE COOLING SYSTEM (Contd)

that 25% of their combined discharge rate is lost from the primary coolant system out the break. The transient hot spot fuel clad temperatures for the break sizes considered are shown on FSAR Figures 14.17.9 to 14.17.13. These demonstrate that the maximum fuel clad temperatures that could occur over the break size spectrum are well below the melting temperature of zirconium (3300°F).

References

(1) FSAR, Section 9.10.3.

(2) FSAR, Section 6.1.

3.4 CONTAINMENT COOLING

Applicability

Applies to the operating status of the containment cooling systems.

Objective

To assure operability of equipment required to remove heat from the containment in normal operating and emergency situations.

Specifications

Containment Cooling Systems

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

a. The following equipment associated with diesel generator 1-2 is operable:

Containment Air Cooler	V1A
Containment Air Cooler	V2A
Containment Air Cooler	V3A
Service Water Pump	P7A
Service Water Pump	P7C
Containment Spray Pump	P54A
Component Cooling Water Pump	P52B

b. The following equipment associated with diesel generator 1-1 is operable:

Containment Air Cooler	V4A
Service Water Pump	P7B
Containment Spray Pump	P54B
Containment Spray Pump	P54C
Component Cooling Water Pump	P52A
Component Cooling Water Pump	P52C

c. All heat exchangers, valves, piping and interlocks associated with the above components and required to function during accident conditions are operable.

3.4.2 During power operation, one of the components listed in Specification 3.4.1 above may be inoperable provided that the corresponding redundant components shall be tested to demonstrate operability. If the inoperable component is not restored to operability within 7 days, the reactor shall be placed in a hot shutdown condition

3.4 CONTAINMENT COOLING (Contd)

within 12 hours. If the inoperable component is not restored to operability within an additional 48 hours, the reactor shall be placed in a cold shutdown condition within 24 hours.

3.4.3 During power operation, the requirements of Specification 3.4.1 may be modified to allow a total of two of the components listed in Section 3.4.1a or b to be inoperable at any one time provided the emergency diesel connected to the opposite engineered safeguards bus is started to demonstrate operability. The redundant component or system on the other bus shall be tested before initiating maintenance on the inoperable components. If the operability of both components is not restored within 24 hours, the reactor shall be placed in a hot shutdown condition within 12 hours. If the operability of both components is not restored within an additional 48 hours, the reactor shall be placed in a cold shutdown condition within 24 hours.

3.4.4 Any valves, interlocks and piping directly associated with one of the above components and required to function during accident conditions shall be deemed to be part of that component and shall meet the same requirements as listed for that component.

3.4.5 Any valve, interlock or piping associated with the containment cooling system which is not covered under Specification 3.4.4 above and which is required to function during accident conditions may be inoperable for a period of no more than 24 hours provided that prior to initiating repairs, all valves and interlocks in the system that provide the duplicate function shall be tested to demonstrate operability.

Basis

A full-capacity emergency diesel generator is connected to each of the two engineered safeguards 2400-volt buses. At least one pump of each type is connected to each of the two buses to assure that equipment is available under all conditions for minimum safety injection. If a pump is removed from one of the two buses for emergency repair, the power supply for minimum

3.4 CONTAINMENT COOLING (Contd)

engineered safety feature equipment is available. It is intended that all equipment be returned to service promptly after emergency repairs are completed. One shutdown heat exchanger is sufficient to cool down the reactor following a shutdown. One heat exchanger is required to remove the heat from the containment after a loss-of-coolant accident. If a heat exchanger is not available, the containment air-cooling system is available as a fully redundant system.

The containment spray system is redundant with the containment air circulation and cooling system.[1] It is sized such that two of the three pumps will limit containment pressure to less than design pressure following a DBA without taking credit for the safety injection tanks.[2] Three of the four air coolers have the capability of limiting the containment pressure under the same conditions as the two pumps.

The redundant equipment provided to limit the containment pressure following a DBA is divided into two groups, each of which has the capacity (with ample reserve) to limit the containment pressure and is connected to a separate engineered safeguards bus. The division of equipment is such that the two spray pumps and one service water pump having full capacity are on one bus and three air coolers and two service water pumps also having full capacity are on the second engineered safeguards bus. In addition, the spare units of these redundant systems are connected to opposite buses to provide excess capacity for pressure reduction on each bus. Therefore, any one unit removed from a given bus for repair does not restrict that group of equipment from completing its design function. The removal of two units from a given bus could limit the capability of that group; therefore, to insure the availability of the power supply to the redundant group in case of outside power failure, the emergency diesel connected to the other bus is started to demonstrate operability.

3.4 CONTAINMENT COOLING (Contd)

During normal power operation, the four fan coolers are in operation to remove heat lost from equipment and piping within containment.[3] The component cooling system components are located in the auxiliary building so as to be accessible for repair after a loss-of-coolant accident.[4]

In addition, if during the post-accident phase the component cooling water supply is lost, core and containment cooling could be maintained until repairs are effected.[5]

References

(1) FSAR, Section 14.18.

(2) FSAR, Section 6.2.3.2.

(3) FSAR, Section 6.3.2.2.

(4) FSAR, Section 9.3.3.2.

(5) FSAR, Section 9.3.2.3.

3.5 STEAM AND FEED-WATER SYSTEMS

Applicability

Applies to the operating status of the steam and feed-water systems.

Objective

To define certain conditions of the steam and feed-water system necessary to assure adequate decay heat removal.

Specifications

3.5.1 The primary coolant shall not be heated above 325°F unless the following conditions are met:

a. Both auxiliary feed-water pumps operable or one auxiliary feed-water pump and one fire pump operable.

b. A minimum of 100,000 gallons of water in the condensate storage and primary coolant system makeup tanks combined and a backup source of additional water from Lake Michigan by the operability of one of the fire protection pumps.

c. All valves, interlocks and piping associated with the above components required to function during accident conditions, are operable.

d. The main steam stop valves are operable and capable of closing in five seconds or less under no-flow conditions.

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 condenser. Therefore, core decay heat can be continuously dissipated via the steam bypass to the condenser as long as feedwater to the steam generator is available. Normally, the capability to supply feedwater to the steam generators is provided by operation of the turbine cycle feed-water system.

In the unlikely event of complete loss of electrical power to the station, decay heat removal is by steam discharge to the atmosphere via the main steam safety valves or power-operated relief valves.[1,2] Either auxiliary feed-water pump can supply sufficient feedwater for removal of decay heat from the plant. The minimum amount of water in the condensate storage tanks is the amount needed for 8 hours of such operation. If the outage is more than 8 hours, Lake Michigan water can be used.

3.5 STEAM AND FEED-WATER SYSTEMS (Contd)

Two fire pumps are provided, one motor-driven and one diesel-driven, each capable of delivering 1500 gpm at 125 psig. In the event all feed pumps should become inoperable, the fire pumps could be used to pump lake water to the steam generators once the secondary pressure was reduced sufficiently by means of the steam dump valve.

A closure time of 5 seconds for the main steam stop valves is considered adequate and was selected as being consistent with expected response time for instrumentation as detailed in the steam line break incident analysis.[3]

References

(1) FSAR, Section 4.3.4.

(2) FSAR, Section 14.13.1.

(3) FSAR, Section 14.14.

3.6 CONTAINMENT SYSTEM

Applicability

Applies to the reactor containment building.

Objective

To assure the integrity of the reactor containment building.

Specifications

3.6.1 Containment Integrity

a. Containment integrity shall not be violated unless the reactor is in the cold shutdown condition.

b. Containment integrity shall not be violated when the reactor vessel head is removed unless the boron concentration is greater than refueling concentration.

c. Except for testing one rod at a time, positive reactivity changes shall not be made by control rod motion or boron dilution unless the containment integrity is intact.

3.6.2 The internal pressure shall not exceed 3 psig (except for containment leak rate tests).

3.6.3 Prior to the reactor going critical after a refueling outage, an administrative check will be made to confirm that all "locked-closed" manual containment isolation valves are closed and locked.

Basis

The primary 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 the primary coolant system ruptures. The shutdown margins are selected based on the type of activities that are being carried out. The refueling boron concentration provides shutdown margin which precludes criticality under any circumstances.

Regarding internal pressure limitations, the containment design pressure of 55 psig would not be exceeded if the internal pressure before a major loss-of-coolant accident were as much as 4 psig.[1]

The containment integrity will be protected if the visual check of all "locked-closed" manual isolation valves to verify them closed is made prior to plant start-up after an extended outage where one or more valves could inadvertently be left open.

References

(1) FSAR, Section 14.18.

3.7 ELECTRICAL SYSTEMS

Applicability

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

Objective

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

3.7.1 Specifications

The primary coolant system shall not be heated or maintained at temperatures above $325^{\circ}F$ if the following electrical systems are not operable:

a. Station power transformer 1-2 (2400 V).

b. Start-up transformer 1-2 (2400 V).

c. 2400 V engineered safeguards buses 1C and 1D.

d. 480 V distribution buses 11 and 12.

e. MCC No 1, 2, 7 and 8.

f. 125 V d-c buses No 1 and 2.

g. Four preferred a-c buses.

h. Two station batteries and the d-c systems including at least one battery charger on each bus.

i. Both diesel generators, with a minimum of 2500 gallons of fuel in each day tank and a minimum of 16,000 gallons of fuel in the underground storage tank.

j. Switchyard battery and the d-c system with one battery charger.

k. 240 V a-c power panels No 1 and 2, and their associated ACB breaker distribution systems.

l. 2400 V bus 1E.

3.7.2 The requirements of Specification 3.7.1 may be modified to the extent that one of the following conditions will be allowed after the reactor has been made critical. If any of the provisions of those exceptions are violated, the reactor shall be placed in a hot shutdown condition within 12 hours. If the violation is not corrected within 24 hours, the reactor shall be placed in a cold shutdown condition within 24 hours.

3.7 ELECTRICAL SYSTEMS (Contd)

a. Station power transformer 1-2 (2400 V) may be inoperable for up to 24 hours provided the operability of both diesel generators is demonstrated immediately.

b. Start-up transformer 1-2 (2400 V) may be inoperable for up to 24 hours provided the operability of both diesel generators is demonstrated immediately. Continued operation beyond 24 hours is permissible provided that a report is sent to the AEC immediately with an outline of the plans for prompt restoration of the start-up transformer and the additional precautions to be taken while the transformer is out of service, and continue operating until notified differently by the AEC.

c. 2400 V engineered safeguards bus 1C or 1D may be inoperable for up to 8 hours provided the operability of the diesel generator associated with the operable bus is demonstrated immediately and there are no inoperable engineered safety feature components associated with the operable bus.

d. 480 V distribution bus 11 or 12 may be inoperable for up to 8 hours provided there are no inoperable safety feature components associated with the operable bus.

e. MCC No 1 and 7 or 2 and 8 may be inoperable for up to 8 hours provided there are no inoperable safety feature components associated with the operable pair of MCC.

f. 125 V d-c bus No 1 or 2 may be inoperable for up to 8 hours provided there are no inoperable safety feature components associated with the operable bus and adequate portable emergency lighting is available during the inoperability of the No 2 bus.

g. One of the four preferred a-c buses may be inoperable for 8 hours provided the reactor protection and engineered safety feature systems supplied by the remaining three buses are all operable.

h. One of the station batteries may be inoperable for 24 hours, providing both battery chargers on the affected bus are in operation.

i. One of the diesel generators may be inoperable for up to 7 days (total for both) during any month, provided the other diesel is started to verify operability, shutdown and the controls are left in the automatic mode, and there are no inoperable engineered safety feature components associated with the operable diesel generator.

3.7 ELECTRICAL SYSTEMS (Contd)

j. 240 V a-c power panel No 1 or power panel No 2 may be inoperable provided that the associated ACB breakers are maintained operable by other means. If the ACB breakers are not maintained operable, either power panel may be inoperable for up to 24 hours provided that the associated ACB breakers are in the "open" position.

k. The switchyard battery may be inoperable for 24 hours provided both battery chargers are operable.

l. The 2400 V bus 1E may be inoperable up to 24 hours.

m. The 125 V d-c power panel No 1 or power panel No 2 may be inoperable provided the associated ACB breakers are maintained operable by other means. If the ACB breakers are not maintained operable, either power panel may be inoperable for up to 24 hours provided that the associated ACB breakers are in the "open" position.

Basis

The electrical system equipment is arranged so that no single contingency can inactivate enough safeguards equipment to jeopardize the plant safety. The 480 V equipment is arranged on two buses. The 2400 V equipment also is supplied from two buses.

The normal source of auxiliary power with the plant at power is from the station power transformers being fed from the main generator with standby power from the start-up transformer and emergency power from either one of two diesel generators.(1) To supplement the standby power source, a spare 345 kV/22 transformer would be available which, in combination with other mobile transformers, could be connected to take the place of the start-up transformer. It is estimated that it could take approximately six weeks to move all of the required equipment on site and to place it in service.(2) There are two emergency power sources on site which do not require outside power or use of the start-up transformer. Upon loss of normal and standby power sources, the 2400 V buses are energized from the diesel generators. Bus load shedding, transfer to the diesel generator and pickup of critical loads are carried out automatically.(3)

When the turbine generator is out of service for an extended period, the generator can be isolated by the removal of links in the bus between the generator and the main transformer, allowing the main transformer and the station power transformers to be returned to service.(1)

3.7 ELECTRICAL SYSTEMS (Contd)

The two 4160 V buses each provide power for two primary coolant pumps and a condensate pump.(4) Operation of the plant could continue if one of these buses were not available as long as power level, etc, were in accordance with Sections 2 and 3.2.2 of these specifications. The design of the electrical system has been carried out with reliability as a prime consideration. Station power is provided from three independent sources. The outage of any two sources will not cause interruption of service to the station power supply 2400 V or below, down to and including 125 V d-c supply.

Equipment served by auxiliary buses and MCC is arranged so that loss of an entire bus does not compromise safety of the plant during DBA conditions.(3) For example, if 2400 V bus 1D is lost, two service water pumps, one containment spray pump and three containment air cooler recirculation fans are lost. This leaves two containment spray pumps, one containment air cooler recirculation fan and one service water pump, which is more than sufficient to control containment pressure below the design value during the DBA.

The requirements for MCC No 1, 2, 7 and 8 and 480 V distributions buses 11 and 12 will assure availability of safety equipment such as charging pumps, boric acid pumps and safety injection valves. The 125 V d-c buses No 1 and No 2 are required for critical instrument and control operations.

The total day tank capacity of 2700 gallons on each diesel is considered more than adequate since approximately 28 hours running time (worst case loading) is available before transfer to fuel oil from the 30,000-gallon underground storage tank is mandatory. Two 20-gpm diesel oil transfer pumps with each being fed from a different diesel are available for transferring fuel oil from the storage tank to the day tanks. In addition, a connection is available outside the diesel rooms to pump oil directly into the day tanks from an oil tanker truck. The 16,000 gallons in the storage tank in addition to the day tank will provide a diesel operation under required loading conditions for a minimum period of 7 days.(5) It is considered incredible not to be able to secure fuel oil from one of several sources within a radius of 70 miles in less than three days under the worst of weather conditions. One battery charger on each battery shall be operating so that the

3.7 ELECTRICAL SYSTEMS (Contd)

batteries will always be at full charge in anticipation of loss-of-ac power incident. This insures that adequate d-c power will be available for starting the emergency generators and other emergency uses. Each battery has two battery chargers available rated at 200 amperes each. Except for the first minute following a DBA, the capacity of the two battery chargers will handle all required loads. The second battery continues to be available in the event of a DBA to pick up the load from its half of the installed engineered safeguards.(6) Each of the reactor protective system channels and the engineered safeguards instrumentation channels is supplied by one of the preferred a-c buses. The removal of one of the preferred a-c buses is permitted as the 2-of-4 logic can be changed to a 2-of-3 or 1-of-2 logic without compromising reactor safety.(7)

Reliable switchyard operation requires the availability of the d-c system in the switchyard for breaker controls and the 240 V a-c system for air compressor operation. The power and air requirements are small, however, and readily adaptable to temporary hookup to alternate power panels or air compressors, so continued operation without the time limitations is permitted provided the affected ACB breakers can be restored to operability through such temporary hookups. In all such situations, restoration to normal conditions shall be accomplished as soon as practicable.

The 2400 V bus 1E is required to assure availability of the pressurizer heaters.

To attain a high degree of reliability for starting and assuming load, the unit will be started to prove operability and shutdown with the controls left in the automatic "start" position.

References

(1) FSAR, Section 8.1.2.
(2) FSAR, Amendment No. 14, Question 8.4.
(3) FSAR, Section 8.3.2.
(4) FSAR, Section 8.3.1.
(5) FSAR, Section 8.4.1.
(6) FSAR, Section 8.4.2.
(7) FSAR, Section 8.3.5.

3.8 REFUELING OPERATIONS

Applicability

Applies to operating limitations during refueling operations.

Objective

To minimize the possibility of an accident occurring during refueling operations that could affect public health and safety.

3.8.1 Specifications

The following conditions shall be satisfied during any refueling operations:

a. One door in each air lock shall be properly closed. In addition, all automatic containment isolation valves shall be operable or at least one valve in each line shall be closed.

b. The containment venting and purge systems, including two radiation monitors that initiate isolation, shall be tested and verified to both be operable immediately prior to refueling operations. The two monitors shall be located on the containment fuel handling area level (elevation 649'), shall be part of the plant area monitoring system, and shall employ one-out-of-two logic for isolation. During normal operation, these monitors will not initiate an isolation signal. A switch shall be provided so that isolation action can be initiated during refueling only.

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

d. Whenever core geometry is being changed, neutron flux shall be continuously monitored by at least two source range neutron monitors, with each monitor providing continuous visual indication in the control room. When core geometry is not being changed, at least one source range neutron monitor shall be in service.

e. At least one shutdown cooling pump and heat exchanger shall be in operation.

3.8 REFUELING OPERATIONS (Contd)

f. During reactor vessel head removal and while refueling operations are being performed in the reactor, the refueling boron concentration shall be maintained in the primary coolant system and shall be checked by sampling on each shift.

g. Direct communication between personnel in the control room and at the refueling machine shall be available whenever changes in core geometry are taking place.

3.8.2 If any of the conditions in 3.8.1 are not met, all refueling operations shall cease immediately, work shall be initiated to satisfy the required conditions, and no operations that may change the reactivity of the core shall be made.

3.8.3 Refueling operation shall not be initiated before the reactor core has decayed for a minimum of 48 hours if the reactor has been operated at power levels in excess of 2% rated power.

3.8.4 The ventilating system and charcoal filter in the fuel storage building shall be operating whenever refueling operations are in process with the equipment door open, or whenever irradiated fuel is being handled in the fuel storage building.

Basis

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

The shutdown margin as 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 250,000 gallons of borated water. The boron concentration of this water

3.8 REFUELING OPERATIONS (Contd)

(1720 ppm boron) is sufficient to maintain the reactor subcritical by approximately 5% $\Delta\rho$ in the cold condition with all rods withdrawn.[2] Periodic checks of refueling water boron concentration insure the proper shutdown margin. Communication requirements allow the control room operator to inform the refueling machine operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

In addition to the above engineered safety features, interlocks are utilized during refueling to insure safe handling. An excess weight interlock is provided on the lifting hoist to prevent movement of more than one fuel assembly at a time. In addition, interlocks on the auxiliary building crane will prevent the trolley from being moved over storage racks containing spent fuel, except as necessary for the handling of fuel.[3] The restriction of not moving fuel in the reactor for a period of 48 hours after the power has been removed from the core takes advantage of the decay of the short half-life fission products and allows any failed fuel to purge itself of fission gases, thus reducing the consequences of a fuel handling accident.

The charcoal filter installed in the fuel handling building exhaust will handle the full 10,500 cfm capacity of the normal ventilation flow with both exhaust fans operating.[4, 5] The normal mode of operation will require that the ventilation supply fan and one exhaust fan be manually tripped following a radioactivity release with a resulting flow of 7300 cfm through the filter. Any radioactivity which should inadvertently, during a refueling operation, pass through the normally opened equipment door would be handled by the charcoal filter in the fuel handling building. The several radiation monitors installed in the containment building and the fuel handling building will give adequate warning to the refueling crew if radioactivity is released. The efficiency of the two installed charcoal filters is at least 90% with rated flows. The off-site thyroid dose in the fuel handling accidents analyzed will be less than 10 Rem using an efficiency

3.8 REFUELING OPERATIONS (Contd)

of 90% and neglecting the scrubbing action of the pool or reactor cavity water should an irradiated fuel bundle be damaged in handling.[5] Valve alignment check sheets are completed to protect against sources of unborated water or draining of the system.

References

(1) FSAR, Section 9.11.

(2) FSAR, Section 3.3.2.

(3) FSAR, Amendment No. 17, Item 13.0.

(4) FSAR, Amendment No. 17, Item 9.0.

(5) FSAR, Section 9.11.2.3.

3.9 EFFLUENT RELEASE

Applicability

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

Objective

To define the conditions for release of radioactive wastes to the circulating water discharge and to the plant vent to prevent off-site doses from exceeding those specified in 10 CFR 20.

Specifications

Liquid Wastes

3.9.1 The release rate of radioactive liquid effluents shall be such that the concentration of radionuclides in the circulating water discharge does not exceed the limits specified in 10 CFR 20, Appendix B, for unrestricted areas on an annual average basis.

3.9.2 The release rates for liquid effluents over any 8-hour period shall not exceed 10 times the limits specified in 10 CFR 20, Appendix B, for unrestricted areas.

3.9.3 Prior to release of liquid waste, a sample shall be taken and analyzed to demonstrate compliance with 3.9.1 above.

3.9.4 During release of liquid radioactivity wastes, the following conditions shall be met:

a. At least one condenser circulating water pump shall be in operation.

b. The gross activity liquids discharge monitor shall be operable.

3.9.5 The minimum holdup time of liquids in the radioactive waste system prior to release to the circulating water discharge shall be as follows:

a. The dirty (high solids, low activity) liquid waste - 2 hours.

b. The clean (low solids, high activity) liquid waste - 2 days.

Gaseous Wastes

3.9.6 The annual average release rates of gaseous and airborne particulate wastes shall be limited in accordance with the following equation:

3.9 EFFLUENT RELEASE (Contd)

$$\sum \frac{Q_i}{(MPC)_i} \leq 5 \times 10^{11} \text{ (cc/sec)}$$

where Q_i is the annual release rate (μCi/sec) of any radioisotope, i, and $(MPC)_i$ in units of μCi/cc are defined in Column 1, Table II, of Appendix B to 10 CFR 20.

3.9.7 The maximum gaseous release rate over any 15-minute period shall not exceed 10 times the yearly average limit.

3.9.8 Prior to release of gaseous wastes from the holding tanks, the contents shall be sampled and analyzed to determine compliance with 3.9.6 and 3.9.7.

3.9.9 For purposes of calculating permissible releases by the above formula, MPC for halogens and particulates with half-lives longer than 8 days will be reduced by a factor of 700 from their listed value in 10 CFR 20, Appendix B.

3.9.10 During release of gaseous wastes to the plant vent, the following conditions shall be met:

a. At least one main exhaust fan shall be in operation.

b. The gaseous radioactivity monitor and the particulate monitor shall be operable during discharges.

3.9.11 During power operation, whenever the air ejector discharge monitor is inoperable, samples shall be taken from the air ejector discharge and analyzed for gross radioactivity daily.

3.9.12 Gaseous radioactive waste shall have the following minimum holdup times prior to release:

a. Potentially high-radioactivity gaseous waste - 7 days (except as noted in 3.9.12c).

b. Low-radioactivity gaseous waste collected by the gas collection header shown on FSAR Figure 11-3.

No holdup required except as provided by piping system itself.

c. Gaseous waste may be discharged from the waste gas surge tank directly to the stack for a period not to exceed 7 days if the holdup system equipment is not available and the release rates meet Specification 3.9.6.

3.9 EFFLUENT RELEASE (Contd)

Basis

Liquid wastes from the radioactive waste disposal system are diluted in the circulating water system discharge prior to release to the lake.[1] With two pumps operating, the rated capacity of the circulating water system is 390,000 gpm. Operation of a single circulating water pump reduces the nominal flow rate by about 50%. The actual circulating water flow under various operating conditions will be calculated from the head differential across the pumps and the manufacturer's head-capacity curves. Because of the low radioactivity levels in the circulating water discharge, the concentrations in the circulating water discharge will be calculated from the measured concentration in the treated waste tank, the flow rate of the treated waste pumps, and the flow in the circulating water system.

If the annual average concentration of liquid wastes in the circulating water discharge equals MPC as specified, the average concentration at the intake of the nearest public water supply at South Haven, Michigan, would be well below MPC.[2, 3] Thus, discharge of liquid wastes at the specified annual average concentrations will not result in significant exposure to members of the public as a result of consumption of drinking water from the lake, even if the effects of potable water treatment systems on reducing radioactive concentration of the water supply is neglected.

The limit for short-term release of liquid wastes is an arbitrary selection based upon reasonable operating practices and will still result in a concentration at the nearest public water intake significantly below MPC.[3] It is intended that the short-term liquid waste release limit normally be restricted to single batches of waste.

The maximum amount of tritium in the discharge is limited to the value given in 10 CFR 20 by imposing a limit on the tritium concentration in the primary coolant water based on minimum dilution (one circulating water pump). As there is no mechanism for concentrating tritium above the concentration in the primary coolant system, there is no safety requirement to monitor the liquid waste discharge for tritium.

The minimum holdup time prior to release of the dirty waste is based on the type of waste (high solids and low radioactivity) and the built-in

3.9 EFFLUENT RELEASE (Contd)

holdup time in the dirty waste system. The low-radioactivity levels in the water processed by this system make holdup unnecessary. Liquids other than the laundry wastes will normally be held up several days. The minimum holdup time prior to release of the clean waste (low solids, high radioactivity) is based upon the processing of these wastes through a filter and two demineralizers in series. The processing itself, as well as the time required for processing, results in the higher radioactivity levels found in these wastes being reduced to levels below MPC. Normally the clean waste liquid will be held in the storage tanks from 7 to 30 days to accomplish additional reduction in radioactivity levels through decay.(4)

Prior to release to the atmosphere, gaseous wastes from the radioactive waste disposal system are mixed with the plant ventilation flow from at least one of two main (60,000 cfm) exhaust fans.(5) Further dilution then occurs in the atmosphere.

The potentially high-radioactivity gaseous waste will be stored in the holdup tanks for at least 7 days to minimize the release of radioactive gas to the environs. Since the compressors are high maintenance items, provision is made to bypass the holdup tanks for a period of 7 days to allow required repairs to equipment. If the gaseous waste cannot be released at levels which satisfy the requirements of 10 CFR 20, Appendix B, the reactor shall be placed in a hot shutdown condition in order to reduce the production of fission gases.

The 15-minute limitation on gaseous release rates in excess of 10 times the annual average limit is based on the fact that radioactivity approximately equivalent to 5% of the annual allowable dose at the site boundary could be released in 15 minutes if the radioactivity level in the primary coolant approached the maximum allowable by Specification 3.1.4.

The low-radioactivity levels associated with the gaseous waste collected by the gas collection header allow it to be piped directly to the base of the plant ventilation tank.

3.9 EFFLUENT RELEASE (Contd)

The formula prescribed in the specification takes atmospheric dilution into account and insures that at the point of maximum ground concentration at the site boundary the requirements of 10 CFR 20 will not be exceeded. The limit is based on the highest long-term value of χ/Q, which occurs at the south edge of the site and is approximately 2×10^{-12} sec/cc. The following data were used to derive the formula prescribed in the specification:

Wind Direction	Average Annual Frequency of Occurrence %	Average Annual Wind Speed m/s	Distance to Site Boundary Downwind m
N	12.51	6.1	677
NNE	4.34	5.5	767
NE	3.46	4.1	Offshore
ENE	3.48	4.1	Offshore
E	4.20	4.4	Offshore
ESE	4.41	4.4	Offshore
SE	5.01	4.5	Offshore
SSE	7.12	5.0	Offshore
S	10.10	5.3	Offshore
SSW	6.21	5.6	811
SW	7.12	6.4	1052
WSW	8.13	6.2	1402
W	7.85	6.8	1315
WNW	6.07	7.9	1227
NW	4.68	6.9	1008
NNW	4.60	6.3	767
Calm	0.66	-	-

3.9 EFFLUENT RELEASE (Contd)

From the FSAR analyses,[6] the following long-term values can be obtained:

Wind Direction	$\frac{Xf}{Q}$ $\frac{sec}{m^3}$	$\frac{Q}{Xf}$ $\frac{m^3}{sec}$
N	2×10^{-6}	5×10^{5}
NNE	5.64×10^{-7}	1.77×10^{6}
SSW	7.39×10^{-7}	1.35×10^{6}
SW	5.23×10^{-7}	1.91×10^{6}
WSW	3.91×10^{-7}	2.56×10^{6}
W	3.88×10^{-7}	2.58×10^{6}
WNW	2.82×10^{-7}	3.54×10^{6}
NW	3.30×10^{-7}	3.03×10^{6}
NNW	5.22×10^{-7}	1.92×10^{6}

Therefore, the highest long-term value of χ/Q at the site boundary is seen to be 2×10^{-6} sec/m^3 or 2×10^{-12} sec/cc.

3.9 EFFLUENT RELEASE (Contd)

References

(1) FSAR, Section 11.1.2.2.

(2) FSAR, Section 2.2.2.

(3) FSAR, Amendment No. 15, Question 2.3.

(4) FSAR, Section 11.1.3.1.

(5) FSAR, Section 9, Figure 9.15.

(6) FSAR, Appendix D.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Applicability

Applies to operation of control rods and hot channel factors during operation.

Objective

To specify limits of control rod movement to assure an acceptable power distribution during power operation, limit worth of individual rods to values analyzed for accident conditions, maintain adequate shutdown margin after a reactor trip and to specify acceptable power limits for power tilt conditions.

Specifications

3.10.1 Shutdown Margin Requirements

a. At full power, the worth of the control rods that are withdrawn shall be at least 2% in reactivity plus the reactivity of the highest worth withdrawn rod plus the reactivity defect of power. At zero power, the worth of the control rods that are withdrawn shall be at least 3.4% in reactivity, plus the reactivity of the highest worth withdrawn rod.

b. For intermediate power level, the regulating group insertion limits as shown on Figure 3-6 shall be maintained to assure that sufficient control rods are withdrawn for the existing power level, including the effect of the highest worth withdrawn rod and the reactivity defect of power.

c. If a control rod cannot be tripped, shutdown margin shall be increased by boration as necessary to compensate for the worth of the withdrawn inoperable rod.

d. The drop time of each control rod shall be no greater than 2.5 seconds from the beginning of rod motion to 90% insertion.

3.10.2 Individual Rod Worth

a. The maximum worth of any one rod in the core shall be equal to or less than 0.43% in reactivity at full power at the beginning of the cycle and 0.50% at the end of the cycle.

b. The maximum worth of any one rod in the core shall be equal to or less than 0.96% in reactivity at zero power at the beginning of the cycle and 1.00% at the end of the cycle.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS (Contd)

3.10.3 Power Distribution Limits

a. If the quadrant to core average power tilt exceeds 15%, except for physics tests, then:

(1) The hot channel factors shall promptly be demonstrated to be less than design values $F_q^N = 3.62$, $F_{\Delta H}^N = 1.94$ or,

(2) Immediate action shall be initiated to reduce reactor power to 75% or less of rated power.

b. If the power in a quadrant exceeds core average by 10% for a period of 24 hours or if the power in a quadrant exceeds core average by 20% at any time, immediate action shall be initiated to reduce reactor power below 50% until the situation is remedied.

c. If the power in a quadrant exceeds the core average by 15%, and if the hot channel factors cannot be demonstrated promptly to be within design limits, then the overpower trip set point shall be reduced to 80% and the thermal margin low-pressure trip set point (P_{trip}) shall be increased by 400 psi.

d. The part-length control rods will be positioned such that the axial power distribution will not produce peaking factors in excess of the design values given above. If the ratio of the power in the upper half to the lower half of the core is not within the range of 0.5 to 1.8 as indicated by the in-core and out-of-core detectors, then immediate action shall be taken to reduce the power below 75% of rated power until the situation is remedied.

3.10.4 Misaligned or Inoperable Control Rod or Part-Length Rod

a. A control rod or a part-length rod is considered misaligned if it is out of position from the remainder of the bank by more than 8 inches.

b. A control rod is considered inoperable if the rod cannot be tripped. No more than one control rod or part length that is either misaligned or inoperable is permitted during power operation.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS (Contd)

c. If a control rod or a part-length rod is misaligned, hot channel factors must promptly be shown to be within design limits or reactor power shall be reduced to 75% or less of rated power within 2 hours. In addition, shutdown margin and individual rod worth limits must be met. Individual rod worth calculations will consider the effects of xenon redistribution and reduced fuel burnup in the region of the misaligned control rod or part-length rod.

3.10.5 Regulating Group Insertion Limits

a. To implement the limits on shutdown margin, individual rod worth and hot channel factors, the limits on control rod regulating group insertion shall be established as shown on Figure 3-6. These limits may be revised during fuel cycle life based on physics calculations and physics data obtained during plant start-up and subsequent operation.

b. The sequence of withdrawal of the regulating groups shall be 1, 2, 3, 4.

c. An overlap of control banks in excess of 40% shall not be permitted.

d. If the reactor is subcritical, the rod position at which criticality could be achieved if the control rods were withdrawn in normal sequence shall not be lower than the insertion limit for zero power shown on Figure 3-6.

3.10.6 Shutdown Rod Limits

a. All shutdown rods shall be withdrawn before any regulating rods are withdrawn.

b. The shutdown rods shall not be withdrawn until normal water level is established in the pressurizer.

c. The shutdown rods shall not be inserted below their exercise limit until all regulating rods are inserted.

Bases

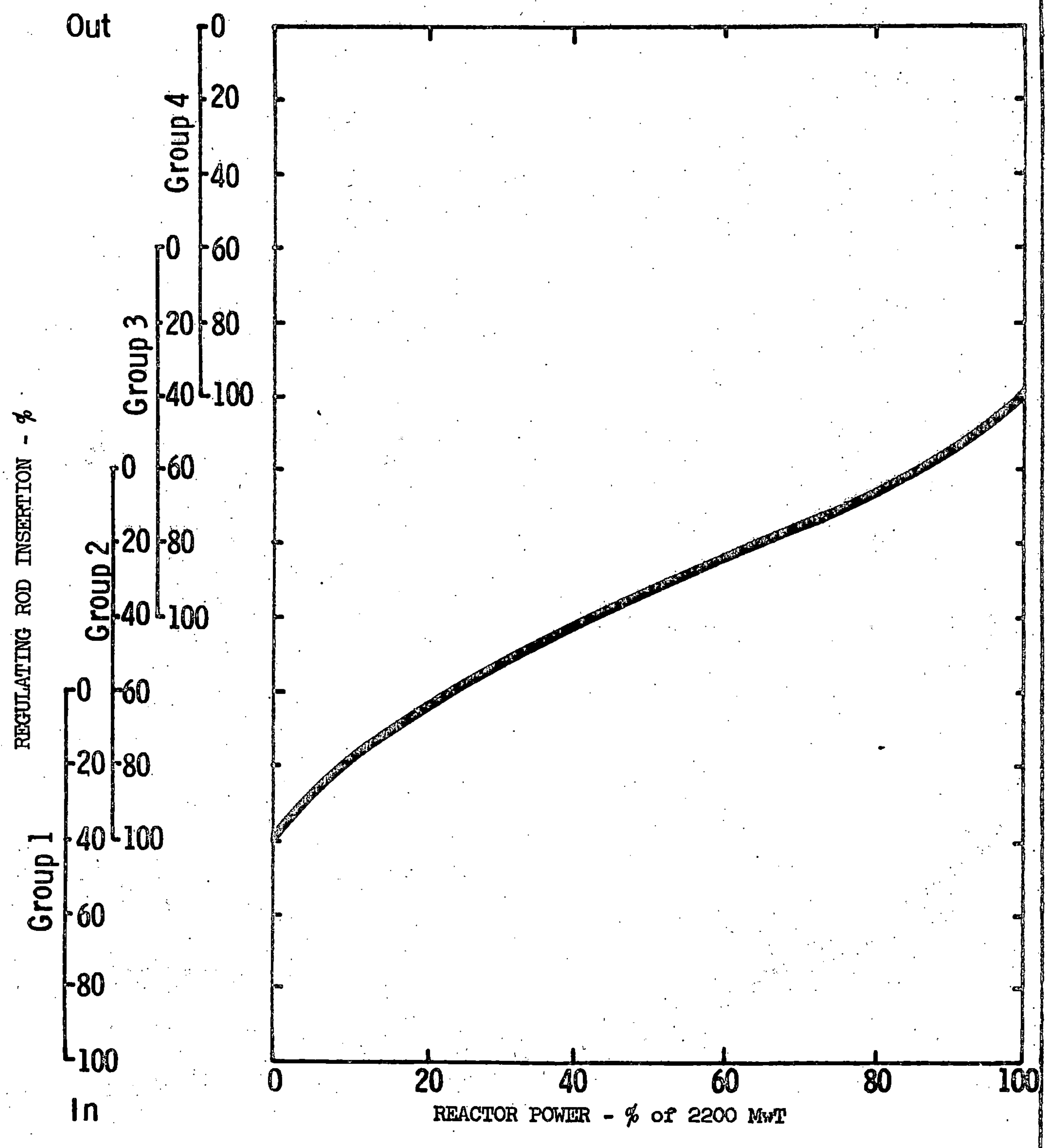
Sufficient control rods shall be withdrawn at all times to assure that the reactivity decrease from a reactor trip provides adequate shutdown margin. The available worth of withdrawn rods must include the reactivity defect of power and the failure of the withdrawn rod

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS (Contd)

of highest worth to insert. The requirement for a shutdown margin of 3.4% in reactivity from a zero power condition and 2.0% in reactivity from a full power condition is consistent with the assumptions used in the analysis of accident conditions as reported in the FSAR.[1] The change in the insertion limit with reactor power shown by Figure 3-6 provides for the necessary increase in withdrawn rod worth to account for the reactivity defect of power and for the increase in worth of the highest worth withdrawn rod. The requirement for 3.4% in reactivity at zero power is based on the value used in the analysis of the postulated steam line break accident at end of fuel cycle life. For earlier fuel cycle lifetimes, this value is conservative. All other analyses of accidents assume a shutdown margin of 2%.

The 2.5 second drop time specified for the control rods is the drop time used in the FSAR safety analysis.[1]

The maximum individual rod worth of inserted control rods and associated peaking factors have been used to demonstrate reactor safety for the unlikely event of a rod ejection accident as described in the FSAR. The maximum worth of an inserted control rod will not exceed the values of the specification for regulating group insertion limits of Figure 3-6. For the regulating group insertion limits of Figure 3-6, the hot channel factors will be well within the design values of $F_q^N = 3.62$ and $F_{\Delta H}^N = 1.94$ used in the FSAR at rated power.[2] Even for a flux tilt (quadrant to core average power of 20%), the hot channel factors will not exceed the design limits. When a flux tilt exists for a sustained time period (24 hours) and cannot be corrected or if a flux tilt reaches 20%, reactor power will be reduced until the tilt can be corrected. A quadrant to core average power tilt may be indicated by three methods: Comparison of the output of the upper or lower sections of the ion chamber with the average value; core outlet thermocouples; and in-core detectors.[3] These values will form the basis for the calculation of peaking factors. Calibration of the out-of-core detectors will take into account the local and total power distribution.



Rod Insertion vs Power Level	Palisades Technical Specifications	Figure 3-6

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS (Contd)

Studies have been performed to show that with this ratio (0.5 - 1.8), total peaking factors are not exceeded. Tests will be performed during start-up testing to verify this relationship.

When part-length control rods are inserted into the active core, they will be positioned as required to control axial zenon oscillations and to maintain peaking factors within the specified limits.

Part-length rod motion will be performed using information from both out-of-core and in-core instrumentation. The relationship of the out-of-core detectors to the power distribution in the core will be verified during start-up testing and subsequent testing and the adequacy of the alarm limits on the Power Ratio Recorder checked.

For a control rod misaligned up to 8 inches from the remainder of the banks, hot channel factors will be well within design limits. If a control rod is misaligned by more than 8 inches, the maximum reactor power will be reduced so that hot channel factors, shutdown margin and ejected rod worth limits are met. If in-core detectors are not available to measure power distribution and rod misalignments >8 inches exist, then reactor power must not exceed 75% to insure that hot channel conditions are met.

For a "dropped" control rod during operation at high power, a turbine runback will automatically decrease the maximum power to 70%.[4] Continued operation with that rod fully inserted will only be permitted if the hot channel factors, shutdown margin and ejected rod worths are satisfied.

In the event a withdrawn control rod cannot be tripped, shutdown margin requirements will be maintained by increasing the boron concentration by an amount equivalent in reactivity to that control rod.

<u>References</u>

(1) FSAR, Section 14.

(2) FSAR, Section 3.3.3.

(3) FSAR, Section 7.4.2.2.

(4) FSAR, Section 7.3.3.6.

3.11 IN-CORE INSTRUMENTATION

Applicability

Applies to the operability of the in-core instrumentation system.

Objective

To specify the functional and operability requirements of the in-core instrumentation system.

Specification

a. Sufficient in-core instrumentation shall be operable whenever the reactor is operating at or above 75% rated power in order to: (1) Assist in the calibration of the out-of-core detectors, and (2) check gross core power distribution. As a minimum, 10 individual detectors per quadrant, which shall include 2 detectors at each of the four axial levels, shall be operable.

b. If the data logger readout for the in-cores is not operable, readings shall be taken and logged on a minimum of 10 individual detectors per quadrant each 2 hours or the reactor power level shall be reduced to less than 75% rated.

Bases

A system of 45 in-core flux detector and thermocouple assemblies and a data display, alarm and record functions has been provided.(1)

The out-of-core nuclear instrumentation calibration includes:

a. Calibration (axial and azimuthal) of the split detectors at initial reactor start-up and during the power escalation program.

b. A comparison check with the in-core instrumentation in the event abnormal readings are observed on the out-of-core detectors during operation.

c. Calibration check during subsequent reactor start-ups.

d. Confirm that the out-of-core axial power splits are as expected and that the ratio of the top and bottom detector readings is acceptable.

Core power distribution verification includes:

a. Measurement at initial reactor start-up to check that power distribution is consistent with calculations.

3.11 IN-CORE INSTRUMENTATION (Contd)

b. Subsequent checks during operation to insure that power distribution is consistent with calculations.

c. Indication of power distribution in the event that abnormal situations occur during reactor operation.

If the data logger for the in-core readout is inoperable, the in-core readings will be manually collected at the terminal blocks in the control room utilizing a suitable signal detector. If this is not feasible with the manpower available, the reactor power will be reduced to less than 75% rated to minimize the probability of exceeding the peaking factors.

The time interval of 2 hours and the minimum of 10 detectors per quadrant are sufficient to maintain adequate surveillance of the core power distribution to detect significant changes until the data logger is returned to service.

Reference

(1) FSAR, Section 7.4.2.4.

3.12 MODERATOR TEMPERATURE COEFFICIENT OF REACTIVITY

Applicability

Applies to the moderator temperature coefficient of reactivity for the core.

Objective

To specify a limit for the positive moderator coefficient.

Specifications

The maximum moderator temperature coefficient at full power conditions shall not exceed $+0.5 \times 10^{-4}$ $\Delta\rho/^{\circ}F$, as obtained by correcting the zero power isothermal measurement to the full power conditions. Uncertainties in the measurement will be added for comparison to the specified limit.

Bases

The maximum positive value of the moderator temperature coefficient used in the accident analyses described in the FSAR was $+0.5 \times 10^{-4} \Delta\rho/^{\circ}F$.(1) Corrections will be applied to the isothermal measurement of the coefficient at zero power to determine if the limiting value at full power (no xenon or samarium) is within the specified limit of $+0.5 \times 10^{-4}$ $\Delta\rho/^{\circ}F$. The following corrections are to be applied to the zero power value:

a. Uncertainty in isothermal measurement:

Considering two possible sources of error, namely the measurement of moderator temperature and control rod position, the moderator temperature coefficient can be measured with a 2 sigma limit of 0.1×10^{-4} $\Delta\rho/^{\circ}F$. In arriving at this accuracy, errors of $2^{\circ}F$ in the temperature measurement and 0.5 inch in rod position have been assumed. Considerable improvement in both of these measurements is possible if analyses indicate that this is warranted.

b. Doppler coefficient at zero power:

The isothermal measurement at zero power must be increased by 0.16×10^{-4} $\Delta\rho/^{\circ}F$ to correct for the Doppler effect present in the measured value.

c. Zero to full power conditions:

To correct for the difference in the interaction of the moderator temperature with the remainder of the system at the two

3.12 MODERATOR TEMPERATURE COEFFICIENT OF REACTIVITY (Contd)

conditions, the zero power measurement should be reduced by 0.11×10^{-4} $\Delta\rho/^{o}F$.

d. Dissolved boron concentration:

This correction is for the difference in boron concentration required between zero and full power. Since more boron is required at zero power, the coefficient is more positive. To correct the zero power measured value to full power, it should be reduced by 0.15×10^{-6} $\Delta\rho/^{o}F$ for each ppm difference in dissolved boron.

e. Moderator temperature:

The moderator temperature increases in going from zero power to full power. As a result, the moderator temperature coefficient is more negative at full power. Thus, the zero power value should be reduced by 0.23×10^{-6} $\Delta\rho/^{o}F$ for each degree difference in moderator temperature from zero power to full power conditions.

f. Azimuthal xexon stability:

A test will be performed during the power test program to verify that divergent azimuthal xenon oscillations do not occur.

Reference

(1) FSAR, Section 14.

3.13 CONTAINMENT BUILDING AND FUEL STORAGE BUILDING CRANES

Applicability

Applies to the use of cranes over the primary coolant system and the spent fuel storage fuel.

Objective

To specify restrictions on the use of the overhead cranes in the Containment Building and the Fuel Storage Building.

Specifications

a. The containment polar crane shall not be used to transport loads over the primary coolant system if the temperature of the coolant or steam in the pressurizer exceeds 225°F.

b. The fuel storage building crane shall not be used to move material past the fuel storage pool unless the crane interlocks are operable or they are bypassed and the crane is under administrative control of a supervisor.

Basis

Loads are not to be allowed over the pressurized primary coolant system to preclude dropping objects which could rupture the boundary of the primary coolant system allowing loss of coolant and overheating of the core.(1)

The fuel storage building crane is provided with a system of trolley and bridge electrical interlocks that will normally prevent the trolley from moving over the storage pool.(2) This minimizes the possibility of dropping an object on the irradiated fuel stored in the pool and resulting in the release of radioactive products. The interlocks may be bypassed under strict administrative control to allow required movement of fuel and material over and to the east of the pool. The crane can be used over the equipment hatches located in the north and south ends of the Fuel Storage Building without the interlocks operable since a load, even if dropped, could not fall into the storage pool.

References

(1) FSAR, Question 2.3.

(2) FSAR, Amendment No. 17, Item 13.

3.14 CONTROL ROOM AIR TEMPERATURE

Applicability

This specification applies to the control room air temperature.

Objective

To limit the temperature under which the reactor protective system and engineered safeguards system instrumentation must function.

Specification

If the control room air temperature reaches 120°F, immediate action shall be taken to reduce this temperature or to place the reactor in a hot shutdown condition.

Basis

The reactor protective system and the engineered safeguards system were designed for and the instrumentation was tested at 120°F. Therefore, if the temperature of the control room exceeds 120°F, the reactor will be shut down and the condition corrected to preclude failure of components in an untested environment.

3.15 REACTOR PRIMARY SHIELD COOLING SYSTEM

Applicability

Applies to the shield cooling system.

Objective

To assure the concrete in the reactor cavity does not overheat and develop excessive thermal stress.

Specification

One shield cooling pump and cooling coil shall be in operation whenever cooling is required to maintain the temperature of the concrete below approximately 165°F.

Basis

The shield cooling system is used to maintain the concrete temperature below 165°F, thus preventing weakening of the structure through loss of moisture. The structure must remain intact during a DBA to preclude damage to the reactor building sump and the plugging of the suction lines to the engineered safeguards pumps. One pump and cooling coil is more than adequate to remove the 120,000 Btu/hr heat load at rated power operation.(1)

Reference

(1) FSAR, Section 9.2.1.

3.16 ENGINEERED SAFETY FEATURES SYSTEM INITIATION INSTRUMENTATION SETTINGS

Applicability

This specification applies to the engineered safety features system initiation instrumentation settings.

Objective

To provide for automatic initiation of the engineered safety features in the event that principal process variable limits are exceeded.

Specifications

The engineered safety features system initiation instrumentation setting limits and permissible bypasses shall be as stated in Table 3.16.1.

Basis

a. High Containment Pressure - The basis for the 5 psig $\begin{Bmatrix}+0.75 \\ -0.25\end{Bmatrix}$ set point for the high-pressure signal is to establish a setting which would be reached immediately in the event of a DBA, cover a spectrum of break sizes and yet be far enough above normal operation maximum internal pressure to prevent spurious initiation.[1,2]

b. Pressurizer Low Pressure - The pressurizer low-pressure safety injection signal is a diverse signal to the high containment pressure safety injection signal. The 1593 psia setting includes an uncertainty of -22 psia and is the setting used in the FSAR Section 14 analysis.[3]

c. Containment High Radiation - Four area monitors in the containment initiate an isolation signal under high radiation condition. The setting is based on the following analysis:

A 10 gpm primary coolant leak to the containment atmosphere is used based upon Specification 3.1.5. Primary coolant radioactivity concentration was assumed to be the maximum allowable by Specification 3.1.4.

(1) Added to this is the contribution from N^{16} whose equilibrium radioactivity in the primary coolant is estimated to be 121 μCi/cc. Semi-infinite cloud geometry and uniform mixing of radioactivity in the containment atmosphere was assumed. N^{16} equilibrium exists in containment atmosphere due to its short

3.16 ENGINEERED SAFETY FEATURES SYSTEM INITIATION INSTRUMENTATION SETTINGS (Contd)

half-life, but all other radioactivity was assumed to build up indefinitely. Calculations show that at the end of a 24-hour leakage period, the dose rate is approximately 20 R/hr as seen by the area monitors. A large leak could exceed the 20 R/hr setting rapidly and initiate isolation.[4]

d. Low Steam Generator Pressure - A signal is provided upon sensing a low pressure in a steam generator to close the main steam isolation valves in order to minimize the temperature reduction in the primary coolant system with resultant loss of water level and possible addition of reactivity. The setting of 500 psia includes a -22 psi uncertainty and was the setting used in the FSAR Section 14 analysis.[5]

e. SIRW Tank Low Level Switches - Level switches are provided on the SIRW tank to actuate the valves in the injection pump suction lines in such a manner so as to switch the water supply from the SIRW tank to the containment sump for a recirculation mode of operation after a period of approximately 20 minutes following a safety injection signal.[5] The switchover point of 27 inches ($^{+0}_{-6}$) above tank bottom is set to prevent the pumps from running dry during the 60 seconds required to stroke the valves and to hold in reserve approximately 20,000 gallons of 1720 ppm borated water. No specific setting was used for the accident analyses stated in the FSAR Section 14.

f. Rod Limit Switches - Power Range Nuclear Instrumentation - Turbine Valve Position Switches - Automatic load reduction to 70% rated power is accomplished by a turbine runback signal to the turbine control valves from dropped rod position switches or from out-of-core nuclear instruments.[9] The settings used for the out-of-core instruments are selected to allow design step changes in power without initiating spurious runback signals.[10] The rate and amount of cutback assure that no reactor core damage will result as the consequence of a dropped rod. The amount of cutback (to 70%) is sufficient to compensate for the maximum calculated single rod worth.[11,12]

3.16 ENGINEERED SAFETY FEATURES SYSTEM INITIATION INSTRUMENTATION SETTINGS (Contd)

g. Engineered Safeguards Pump Room Vent-Radiation Monitor - A process monitor is installed to provide an isolation signal upon high radioactivity levels in the engineered safeguards pump rooms. The setting is based on the following analyses:

To maintain acceptable dose levels at the site boundary, it is necessary not only to detect significant quantities of leakage into the east and west engineered safeguards pump rooms, but also to provide a reasonable trip point at which ventilation of the rooms will terminate. For this, the following analysis was performed: Primary coolant radioactivity concentration was assumed to be the maximum allowable by Specification 3.1.4. No fuel melting is assumed to occur. An average beta energy was calculated for each nuclide for purposes of converting individual isotopic radioactivity concentrations seen by the process monitor to count rates measured by the monitor. The design exhaust ventilation rate of 2400 CFM was assumed along with a 1 gpm leak rate into the room. This leakage is made up of primary coolant (81,800 gallons) diluted with up to 285,000 gallons of SIRW tank water and 7,480 gallons of safety injection tank water. This results in a total primary coolant radioactivity minus noble gases residing in 374,280 gallons of water. The results show that the process monitor registered a count rate of about 2.2×10^5 cpm. Normal background for this monitor is expected to be less than 1×10^3 cpm. Furthermore, due to the wide variation between normally expected background count rate and the count rate registered during a 1 gpm leak rate, detection of far smaller leak rates can be expected. This is especially true in the event of fuel meltdown. The relative safety implications of a 1 gpm leak rate into the engineered safeguards pump room are minor (per basis of Specification 3.1.5).[4]

3.16 ENGINEERED SAFETY FEATURES SYSTEM INITIATION INSTRUMENTATION SETTINGS (Contd)

References

(1) FSAR, Amendment No. 16, Question 6.4.

(2) FSAR, Amendment No. 17, Item 4.

(3) FSAR, Section 14.14.3.

(4) FSAR, Section 11.

(5) FSAR, Section 14.14.2.

(6) FSAR, Section 6.1.2.2.

(7) FSAR, Section 7.3.1.3.

(8) FSAR, Section 14.12.

(9) FSAR, Section 7.3.2.6.

(10) FSAR, Section 7.3.3.6.

(11) FSAR, Section 14.4.

(12) FSAR, Amendment No. 15, Item 7.4.

Table 3.16.1

Engineered Safety Features System Initiation Instrument Setting Limits

Functional Unit	Channel	Setting Limit
1. High Containment Pressure	a. Safety Injection b. Containment Spray c. Containment Isolation d. Containment Air Cooler DBA Mode	≤ 5 Psig $\left(\begin{smallmatrix}+0.75\\-0.25\end{smallmatrix}\right.$
2. Pressurizer Low Pressure	Safety Injection	≥ 1593 Psia[(1)]
3. Containment High Radiation	Containment Isolation	≤ 20 R/Hr
4. Low Steam Generator Pressure	Steam Line Isolation	≥ 500 Psia[(2)]
5. SIRW Low Level Switches	Recirculation Actuation	≤ 27-Inch $\left(\begin{smallmatrix}+0\\-6\end{smallmatrix}\right.$ Above Tank Bottom
6. Rod Limit Switches (LS-6)	Turbine Cutback	≤ 5 Inches
7. Power Range Nuclear Instr	Turbine Cutback	a. Time Delay 8 Sec $\left(\begin{smallmatrix}+1\\-1\end{smallmatrix}\right.$ b. $\leq 8\%$ Power
8. Turbine Valve Position Switches	Turbine Cutback	$\leq 70\%$ Rated Power
9. Engineered Safeguards Pump Room Vent - Radiation Monitors	Engineered Safeguards Pump Room Isolation	$\leq 2.2 \times 10^5$ Cpm

(1) May be bypassed below 1700 psia and is automatically reinstated above 1700 psia.
(2) May be bypassed below 550 psia and is automatically reinstated above 550 psia.

3.17 INSTRUMENTATION AND CONTROL SYSTEMS

Applicability

Applies to plant instrumentation systems.

Objective

To delineate the conditions of the plant instrumentation and control systems necessary to assure reactor safety.

Specifications

3.17.1 The operability of the plant instrument and control systems shall be in accordance with Tables 3.17.1 through 3.17.4.

3.17.2 In the event the number of channels of a particular system in service falls below the limits given in the columns entitled "Minimum Operable Channels" or "Minimum Degree of Redundancy," except as conditioned by the column entitled "Permissible Bypass Conditions," the reactor shall be placed in a hot shutdown condition within 12 hours. If minimum conditions are not met within 24 hours, the reactor shall be placed in a cold shutdown condition within 24 hours.

Basis

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 and engineered safeguards channels are supplied with sufficient redundancy to provide the capability for channel test at power. Exceptions are backup channels such as loss-of-load trip.

When one of the four channels is taken out of service for maintenance, the protective system logic can be changed to a two-out-of-three coincidence for a reactor trip by bypassing the removed channel.

3.17 INSTRUMENTATION AND CONTROL SYSTEMS (Contd)

If the bypass is not effected, the out-of-service channel (Power Removed) assumes a tripped condition (except high rate-of-change of power, high power level and high pressurizer pressure),[1] which results in a one-out-of-three channel logic. If, in the 2 of 4 logic system of either the reactor protective system or the engineered safeguards system, one channel is bypassed and a second channel manually placed in a tripped condition, the resulting logic is 1 of 2. At rated power, the minimum operable high-power level channels is 3 in order to provide adequate flux tilt detection. If only 2 channels are operable, the reactor power level is reduced to 70% rated power which protects the reactor from possibly exceeding design peaking factors due to undetected flux tilts and from exceeding dropped rod peaking factors in the event that a turbine runback signal is required from the power range channels.

The engineered safeguards system provides a 2 of 4 logic on the signal used to actuate the equipment connected to each of the 2 emergency diesel generator units.

References

(1) FSAR, Section 7.2.7.

(2) FSAR, Section 7.2.2.

Table 3.17.1

Instrumentation Operating Requirements for Reactor Protective System

No	Functional Unit	Minimum Operable Channels	Minimum Degree of Redundancy	Permissible Bypass Conditions
1	Manual (Trip Buttons)	1	None	None
2	High-Power Level	(2)(b,d)	(1)(d)	None
3	Log Range Channels	2	1	Below 10^{-4}% or Above 15% Rated Power(a) Except as Noted in (c)
4	Thermal Margin/ Low-Pressurizer Pressure	2(b)	1	Below 10^{-4}% of Rated Power(a)
5	High-Pressurizer Pressure	2(b)	1	None
6	Low Flow Loop	2(b)	1	Below 10^{-4}% of Rated Power(a)
7	Loss of Load	1	None	None
8	Low Steam Generator Water Level	2/Steam Gen(b)	1/Steam Generator	None
9	Low Steam Generator Pressure	2/Steam Gen(b)	1/Steam Generator	Below 10^{-4}% of Rated Power(a)
10	High Containment Pressure	2(b)	1	None

(a) Bypass automatically removed.
(b) One of the inoperable channels must be in the tripped condition.
(c) Two channels required if TM/LP, low steam generator or low-flow channels are bypassed.
(d) If only two channels or less are operable, load shall be reduced to 70% or less of rated power.

Table 3.17.2

Instrumentation Operating Requirements for Engineered Safety Feature Systems

No	Functional Unit	Minimum Operable Channels	Minimum Degree of Redundancy	Permissible Bypass Conditions
1	Safety Injection			
a.	Manual (Trip Buttons)	1	None	None
b.	High Containment Pressure	2[a,c]	1	During Leak Test
c.	Pressurizer Low Pressure	2[c]	1	Primary Pressure Less Than 1700 Psia[b]
2	Containment Spray			
a.	Manual	2	None	None
b.	High Containment Pressure	2[a,c]	1	During Leak Test

(a) Right and left actuation circuits each have 2 channels.
(b) Auto removal of bypass above 1700 psia.
(c) One of the inoperable channels must be in the tripped condition.

Table 3.17.3

Instrument Operating Conditions for Isolation Functions

No	Functional Unit	Minimum Operable Channels	Minimum Degree of Redundancy	Permissible Bypass Conditions
1	Containment Isolation			
a.	Containment High Pressure	2[a,c]	1	During Leak Test
b.	Containment High Radiation	2[c]	1	None
c.	Manual	1	None	None
2	Steam Line Isolation			
a.	Low Steam Gen Pressure	2/Steam Gen[c]	1	Below 550 Psia[b]
b.	Manual	1/Steam Gen	None	None

(a) Right and left actuation circuits each have 2 channels.
(b) Bypass automatically reinstated above 550 psia.
(c) One of the inoperable channels must be in the tripped position.

Table 3.17.4

Instrumentation Operating Requirements for Other Safety Feature Functions

No	Functional Unit	Minimum Operable Channels	Minimum Degree of Redundancy	Permissible Bypass Conditions
1	SIRW Low-Level Switches	2[(b)]	1	None
2	ΔT - Power Comparator	3[(c)]	1	None
3	High-Pressure Safety Injection Flow Instruments	4	None	None
4	Air Cooler Service Water Flow Instr	1	None	None
5	Primary and Secondary Rod Insertion and Out-of-Sequence Monitors	1	None	NA
6	Fuel Pool Bldg Crane Interlocks	1	None	As Requested Under Administrative Control[(a)]
7	Start-Up Channels	2	1	Not Required Above $10^{-4}\%$ of Rated Power

(a) Crane shall not be used to move material past the fuel storage pool unless the interlocks are available.

(b) One of the inoperable channels must be in the tripped condition.

(c) If only 2 channels are operable, load shall be reduced to 70% or less of rated power.

4.0 SURVEILLANCE REQUIREMENTS

Unless otherwise specified, intervals may be adjusted plus or minus 25% to accommodate normal test schedules.

4.1 INSTRUMENTATION AND CONTROL

Applicability

Applies to the reactor protective system and other critical instrumentation and controls.

Objective

To specify the minimum frequency and type of surveillance to be applied to critical plant instrumentation and controls.

Specifications

Calibration, testing, and checking of instrument channels, reactor protective system and engineered safeguards system logic channels and miscellaneous instrument systems and controls shall be performed as specified in Tables 4.1.1 to 4.1.3.

Basis

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

Based on experience in operation of both conventional and nuclear plant systems when the plant is in operation, a checking frequency of once-per-shift is deemed adequate for reactor and steam system instrumentation. Calibrations are performed to insure the presentation and acquisition of accurate information.

The power range safety 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 each refueling shutdown interval.

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 power range safety channels, and once each refueling shutdown for the process system channels, are considered adequate.

The minimum testing frequency for those instrument channels connected to the reactor protective system is based on an estimated average unsafe failure rate of 1.14×10^{-5} failure/hour per channel. This estimation is based on limited 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 the specified one-month test interval, the average unprotected time is 360 hours in case of a failure occurring between test intervals, thus the probability of failure of one channel between test intervals is $360 \times 1.14 \times 10^{-5}$ or 4.1×10^{-3}. Since two channels must fail in order to negate the safety function, the probability of simultaneous failure of two-out-of-four channels is $(4.1 \times 10^{-3})^3 = 6.9 \times 10^{-8}$. This represents the fraction of time in which each four-channel system would have one operable and three inoperable channels and equals $6.9 \times 10^{-8} \times 8760$ hours per year, or 2.16 seconds/year.

These estimates are conservative and may be considered upper limits. Testing intervals will be adjusted as appropriate based on the accumulation of specific operating history.

The testing frequency of the process instrumentation is considered adequate (based on experience at other conventional and nuclear plants on Consumers Power Company's system) to maintain the status of the instruments so as to assure safe operation.

Those instruments which are similar to the reactor protective system instruments are tested at a similar frequency and on the same basis.

Table 4.1.1

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF REACTOR PROTECTIVE SYSTEM

Channel Description	Surveillance Function	Frequency	Surveillance Method
1. Power Range Safety Channels	a. Check	S	a. Comparison of four power channel readings.
	b. Calibrate[3]	D	b. Channel adjustment to agree with heat balance calculation. Repeat whenever flux-ΔT power comparator alarms.
	c. Test	M[2]	c. Internal test signal to verify trips, alarms, turbine runback signal, permissives.
2. Wide-Range Logarithmic Neutron Monitors	a. Check	S	a. Comparison of both wide-range readings.
	b. Test	P	b. Internal test signals to verify SUR indication and trip, power level permissives, instrument accuracy.
3. Reactor Coolant Flow	a. Check	S	a. Comparison of four separate total flow indications.
	b. Calibrate	R	b. Known differential pressure applied to sensors to calibrate all loop devices.
	c. Test	M[2]	c. Bistable trip tester.[1]
4. Thermal Margin/Low Pressurizer Pressure	a. Check: (1) Temperature Input	S	a. Check: (1) Comparison of four separate calculated trip pressure set point indications.
	(2) Pressure Input		(2) Comparison of four pressurizer pressure indications. (Same as 5(a) below.)

Table 4.1.1

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF REACTOR PROTECTIVE SYSTEM (Contd)

Channel Description	Surveillance Function	Frequency	Surveillance Method
4. (Contd)	b. Calibrate: (1) Temperature Input	R	b. Calibrate: (1) Known resistance substituted for RTD coincident with known pressure input.
	(2) Pressure Input		(2) Known pressure applied to sensor coincident with above temperature calibrations.
	c. Test	M(2)	c. Bistable trip tester.(1)
5. High-Pressurizer Pressure	a. Check	S	a. Comparison of four separate pressure indications.
	b. Calibrate	R	b. Known pressure applied to sensors.
	c. Test	M(2)	c. Bistable trip tester.(1)
6. Steam Generator Level	a. Check	S	a. Comparison of four level indications per generator.
	b. Calibrate	R	b. Known differential pressure applied to sensors.
	c. Test	M(2)	c. Bistable trip tester.(1)
7. Steam Generator Pressure	a. Check	S	a. Comparison of four pressure indications per generator.
	b. Calibrate	R	b. Known pressure applied to sensors.
	c. Test	M(2)	c. Bistable trip tester.(1)
8. Containment Pressure	a. Calibrate	R	a. Known pressure applied to sensors.
	b. Test	M(2)	b. Simulate pressure switch action.
9. Loss of Load	a. Test	P	a. Manually trip turbine auto stop oil relays.

Table 4.1.1

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF REACTOR PROTECTIVE SYSTEM (contd)

Channel Description	Surveillance Function	Frequency	Surveillance Method
10. Manual Trips	a. Test	P	a. Manually test both circuits.
11. Reactor Protection System Logic Units	a. Test	M(2)	a. Internal test circuits check logic networks and clutch power relays.

Notes: (1) The bistable trip tester injects a signal into the bistable and provides a precision readout of the trip set point.

(2) All monthly tests will be done on only one of four channels at a time to prevent reactor trip.

(3) Calibrate using built-in simulated signals.

S - Each Shift
D - Daily
M - Monthly
R - Each Refueling Shutdown
P - Prior to Each Start-Up if Not Done Previous Week

Table 4.1.2

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF ENGINEERED SAFETY FEATURE INSTRUMENTATION CONTROLS

Channel Description	Surveillance Function	Frequency	Surveillance Method
1. Low-Pressure SIS Initiation Channels	a. Check	S	a. Comparison of four separate pressure indications.
	b. Calibrate	R	b. Known pressure applied to sensors and SIS actuation logic verified.
	c. Test	M	c. Signal to meter relay adjusted with test device to trip one channel at a time.
2. Low-Pressure SIS Signal Block Permissive and Auto Reset	a. Calibrate	R	a. Part of 1(b) above.
3. SIS Actuation Relays	a. Test	Q	a. Simulation of SIS 2/4 logic trip using built-in testing system. Both "standby power" and "no standby power" circuits will be tested for left and right channels. Test will verify functioning of initiation circuits of all equipment normally operated by SIS signals.
	b. Test	R	b. Complete automatic test initiated by sensor operation (Item 1(b)) and including all normal automatic operations.
4. Containment High-Pressure Channels	a. Calibrate	R	a. Known pressure applied to sensors and CHP actuation logic for SIS, containment isolation and containment spray verified.

Table 4.1.2

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF ENGINEERED SAFETY FEATURE INSTRUMENTATION CONTROLS (Contd)

Channel Description	Surveillance Function	Frequency	Surveillance Method
4. (Contd)	b. Test	M	b. Pressure switch operation simulated by opening or shorting terminals, one circuit at a time.
5. Containment High Radiation Channels	a. Check	D	a. Comparison of four separate radiation level indications.
	b. Calibrate	R	b. Exposure to known external radiation source.
	c. Test	M	c. Remote operated integral radiation check source used to verify instrument operation, one channel at a time.
	d. Test	R	d. Simulation of CHR 2/4 logic trip with test switch to verify actuation relays, including containment isolation.
6. Manual SIS Initiation	a. Test	R	a. Manual push-button test.
7. Manual Containment Isolation Initiation	a. Test	R	a. Manual push-button test.
	b. Check	R	b. Observe isolation valves closure.
8. Manual Initiation Containment Spray Pumps and Valves	a. Test	R	a. Manual switch operation.
9. DBA Sequencers	a. Test	Q	a. Proper operation will be verified during SIS actuation test of Item 3(a) above.

Table 4.1.2

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF ENGINEERED SAFETY FEATURE INSTRUMENTATION CONTROLS (Contd)

Channel Description	Surveillance Function	Frequency	Surveillance Method
10. Normal Shutdown Sequencers	a. Test	R	a. Simulate normal actuation with test-operate switch and verify equipment starting circuits.
11. Diesel Start	a. Test	M	a. Manual initiation followed by synchronizing and loading.
	b. Test	Q	b. Diesels will be started during SIS actuation test of 3(a) above.
	c. Test	R	c. Diesel start, load shed, synchronizing, and loading will be verified during Item 3(b) above.
	d. Test	P	d. Diesel auto start initiating circuits.
12. SIRW Tank Level Switch Interlocks	a. Test	R	a. Level switches removed from fluid to verify actuation of valves.
	b. Test	Q	b. Use SIRW tank control switch.
13. Safety Injection Tank Level and Pressure Instruments	a. Check	S	a. Verify that level and pressure indication is between independent high high/low alarms for level and pressure.
	b. Calibrate	R	b. Known pressure and differential pressure applied to pressure and level sensors.
14. Boric Acid Tank Level Switches	a. Test	R	a. Pump tank below low-level alarm point to verify switch operation.

Table 4.1.2

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF ENGINEERED SAFETY FEATURE INSTRUMENTATION CONTROLS (Contd)

Channel Description	Surveillance Function	Frequency	Surveillance Method
15. Boric Acid Heat Tracing System	a. Check	D	a. Observe temperature recorders for proper readings.
16. Main Steam Isolation Valve Circuits	a. Check	S	a. Compare four independent pressure indications.
	b. Calibrate	R	b. Known pressure applied to sensors to verify trip points, logic operation, block permissive, auto reset and valve closures.
17. SIRW Tank Temperature Indication and Alarms	a. Check	M	a. Compare independent temperature readouts.
	b. Calibrate	R	b. Known resistance applied to indicating loop from the sensor and alarm functionally checked.

S - Each Shift
D - Daily
M - Monthly
Q - Quarterly
R - Each Refueling Shutdown
P - Prior to Each Start-Up if not Done Previous Week

Table 4.1.3

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS, AND TESTING OF MISCELLANEOUS INSTRUMENTATION AND CONTROLS

Channel Description	Surveillance Function	Frequency	Surveillance Method
1. Startup Range Neutron Monitors	a. Check	S	a. Comparison of both channel countrate indications when in service.
	b. Test	P	b. Internal test signals used to verify instrument accuracy and SUR alarm.
2. Primary Rod Position Indication System	a. Check	S	a. Comparison of output data with secondary RPIS.
	b. Check	M	b. Check of power dependent insertion limits monitoring system.
	c. Calibrate	R	c. Physically measured rod drive position used to verify system accuracy. Check rod position interlocks.
3. Secondary Rod Position Indication System	a. Check	S	a. Comparison of output data with primary RPIS.
	b. Check	M	b. Same as 2 (b) above.
	c. Calibrate	R	c. Same as 2 (c) above, including out of sequence alarm function.
4. Area and Process Monitors	a. Check	D	a. Normal readings observed and internal test signals used to verify instrument operation.
	b. Calibrate	R	b. Exposure to known external radiation source.
	c. Test	M	c. Detector exposed to remote operated radiation check source.

Table 4.1.3

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS, AND TESTING OF MISCELLANEOUS INSTRUMENTATION AND CONTROLS (Contd)

Channel Description	Surveillance Function	Frequency	Surveillance Method
5. Emergency Plan Radiation Instruments	a. Calibrate	A	a. Exposure to known radiation source.
	b. Test	M	b. Battery Check.
6. Environmental Monitors	a. Check	M	a. Operational check.
	b. Calibrate	A	b. Verify airflow indicator.
7. Pressurizer Level Instruments	a. Check	S	a. Comparison of six independent level readings.
	b. Calibrate	R	b. Known differential pressure applied to sensor.
	c. Test	M	c. Signal to alarm meter relay adjusted with test device to verify setting.
8. Control Rod Drive System Interlocks	a. Test	R	a. Verify proper operation of all rod drive control system interlocks, using simulated signals where necessary.
	b. Test	P	b. If haven't been checked for three months and plant is shutdown.
9. Turbine Runback	a. Test	M	a. Check combination nuclear instrumentation and rod drive control system signal with test circuit.
	b. Test	R	b. Insert rod drives below lower electrical limit to verify runback signal initiation.

Table 4.1.3

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS, AND TESTING OF MISCELLANEOUS INSTRUMENTATION AND CONTROLS (Contd)

Channel Description	Surveillance Function	Frequency	Surveillance Method
10. Flux-ΔT Power Comparator	a. Calibrate	R	a. Use simulated signals.
	b. Test	M	b. Use built-in test devices.
11. Calorimetric Instrumentation	a. Calibrate	R	a. Apply known d/p to feedwater flow sensors.
12. Containment Building Humidity Detectors	a. Test	R	a. Place sensor in high humidity atmosphere.
13. Interlocks - Isolation Valves on Shutdown Cooling Line	a. Calibrate	R	a. Known differential pressure of 300 psi applied across disc and torque switch adjusted.
14. Service Water Break Detector in Containment	a. Test	R	a. Apply test signals to flow transmitters and actuate flow switch and alarm.
15. Control Room Ventilation	a. Test	R	a. Check damper operation for DBA mode with HS1801 and isolation signal.
	b. Test	R	b. Check control room for positive pressure.

S - Each Shift
D - Daily
M - Monthly
A - Annually
R - Each Refueling Shutdown
P - Prior to Each Startup

4.2 EQUIPMENT AND SAMPLING TESTS

Applicability

Applies to plant equipment and conditions related to safety.

Objective

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

Specifications

Equipment and sampling tests shall be conducted as specified in Tables 4.2.1 and 4.2.2.

Basis

Sampling and Equipment Testing

The equipment testing and system sampling frequencies specified in Tables 4.2.1 and 4.2.2 are considered adequate based upon experience, to maintain the status of the equipment and systems so as to assure safe operation. Thus, those systems where changes might occur relatively rapidly are sampled frequently; and those static systems not subject to changes are sampled less frequently.

Table 4.2.1

Minimum Frequencies for Sampling Tests

		Test	Frequency	FSAR Section Reference
1.	Reactor Coolant Samples	Gross β, γ Radioactivity and Qualitative Gamma Spectral Analyses	3 Times/Week(1)	None
		Tritium Radioactivity	Weekly	None
		Chemistry (Cl and O_2)	3 Times/Week	None
		Radiochemical Analysis for $\bar{E}$ Determination	Semiannual(2)	None
2.	Reactor Coolant Boron	Boron Concentration	Twice/Week	None
3.	SIRW Tank Water Sample	Boron Concentration	Monthly	None
4.	Concentrated Boric Acid Tanks	Boron Concentration	Monthly	None
5.	SI Tanks	Boron Concentration	Monthly	6.1.2
6.	Spent Fuel Pool	Boron Concentration	Monthly	9.4
7.	Secondary Coolant	Iodine Concentration	Weekly(3)	None
8.	Liquid Radwaste	Radioactivity Analysis	Prior to Release of Each Batch	11.1
9.	Radioactive Gas Decay Tanks	Radioactivity Analysis	Prior to Release of Each Batch	11.1
10.	Stack-Gas Monitor Particulate Samples	Iodine 131 and Particulate Radioactivity	Weekly(4)	11.1

(1) When radioactivity level exceeds 10 percent of limits in Specification 3.1.4, or 3.1.5, the sampling frequency shall be increased to a minimum of once each day.

(2) Redetermined if the primary coolant radioactivity changes by more than 10 μCi/cc in accordance with Specification 3.1.4.

(3) When radioactivity level exceeds 10 percent of limits in Specification 3.1.5(b), the sampling frequency shall be increased to a minimum of once each day.

(4) When iodine or particulate radioactivity levels exceed 10 percent of limit in Specification 3.9.6 and 3.9.9, the sampling frequency shall be increased to a minimum of once each day.

Table 4.2.2

Minimum Frequencies for Equipment Tests

		Test	Frequency	FSAR Section Reference
1.	Control Rods	Drop Times of All Full-Length Rods	Each Refueling Shutdown	7.4.1.3
2.	Control Rods	Partial Movement of All Rods (Minimum of 6 In)	Every Two Weeks	7.4.1.3
3.	Pressurizer Safety Valves	Set Point	One Each Refueling Shutdown	7.3.7
4.	Main Steam Safety Valves	Set Point	Five Each Refueling Shutdown	4.3.4
5.	Refueling System Interlocks	Functioning	Prior to Refueling Operations	9.11.3
6.	Service Water System Valve Actuation (SIS-CHP)	Functioning	Each Refueling Operation	9.1.2
7.	Fire Protection Pumps and Power Supply	Functioning	Monthly	9.6.2
8.	Primary System Leakage	Evaluate	Daily	4 Amend 15, Ques 4.3.7
9.	Diesel Fuel Supply	Fuel Inventory	Daily	8.4.1
10.	Critical Headers Service Water System	150 Psig Hydrostatic Test	Every Five Years	9.1.2
11.	Charcoal & Hi Efficiency Filters for Control Room Fuel Storage Building and Containment H_2 Purge System	Charcoal and Absolute Filter Efficiencies Checked $\geq$ 99% for Iodine and 0.3 Micron Particulate. DOP Test on Absolute Filters. Freon Test on Charcoal Filter Units.	Each Refueling Shutdown and at any time work on filters could alter their integrity.	Amend 14, Ques 14.19-1 6.5.1 9.8.3

4.3 PRIMARY SYSTEM SURVEILLANCE

Applicability

Applies to preoperational and in-service structural surveillance of the reactor vessel and other primary system components.

Objective

To insure the integrity of the primary coolant system.

Specifications

a. Prior to initial plant operation, an ultrasonic survey shall be made of reactor vessel shell welds, vessel nozzles, vessel flange welds, piping system butt welds and major welds on the pressurizer and steam generators to establish preoperational system integrity and basic conditions for future testing.

b. Postoperational inspections shall be made according to the methods and intervals indicated in Table 4.3.1 and IS 242 of Section XI of the ASME B&PV Code.

c. The structural integrity of the primary system boundary shall be maintained at the level required by the original acceptance standards throughout the life of the plant. Any evidence, as a result of the tests outlined in Table 4.3.1, that defects have developed or grown shall be investigated, including evaluation of comparable areas of the primary system.

d. Sufficient records of each inspection shall be kept to allow comparison and evaluation of future tests.

e. The in-service inspection program shall be reevaluated at the end of five years to consider incorporation of new inspection techniques that have been proven practical and the conclusions of this evaluation shall be reviewed with the AEC.

f. Surveillance of the regenerative heat exchanger and primary coolant pump flywheels shall be performed as indicated in Table 4.3.2.

g. A surveillance program to monitor radiation-induced changes in the mechanical and impact properties of the reactor vessel materials shall be maintained as described in Section 4.5.3 of the FSAR. The specimen removal schedule shall be as indicated in Table 4.3.3.

Basis

The inspection program specified, places major emphasis on the areas of highest stress concentration as determined by general design evaluation

4.3 PRIMARY SYSTEM SURVEILLANCE (Contd)

and experience with similar systems.[1] In addition, that portion of the reactor vessel shell welds which will be subjected to a fast neutron dose sufficient to change ductility properties will be inspected. The inspections will rely primarily on ultrasonic methods utilizing up-to-date analyzing equipment and trained personnel. To the extent applicable, the testing techniques and acceptance criteria of Section XI of the ASME B&PV Code will be utilized. Preoperational inspections will establish base conditions by determining indications that might occur from geometrical or metallurgical sources and reading from discontinuities in weldments or plates which might cause undue concern on a postservice inspection. Table 4.3.1 duplicates Table IS 261 of Section XI of the code for ease of comparison. To the extent applicable, based upon the existing design and construction of the plant, Table IS 261 has been followed. Significant exceptions are as follows:

Item 1.2 - Reference to meridional and circumferential seam welds in bottom head and closure head has been deleted since the closure head is inaccessible to volumetric equipment other than at the flange; and since the techniques and equipment for inspecting the bottom head are not available.

Item 1.5 - This item has been deleted since the plant design and present technology do not permit a volumetric examination of this type of penetration.

Item 1.7 - The transition pieces between the carbon steel nozzles and the carbon steel piping are also carbon steel and thus not "safe ends." The examination category "J" which applies to piping also applies to these welds rather than category "F." Also, the requirement for surface examination has been deleted since these areas are inaccessible. Since the plant design calls for examination from inside the reactor vessel, the "visual" requirement is not possible.

Items 1.11 and 1.12 are not applicable since no bolting 2-inch and over is used, nor are any integrally welded reactor vessel supports.

Item 1.15 - Reference to integrally welded internal supports is deleted since none exist. The bottom snubbers and drop prevention supports normally carry no load and do not fit this category. These items will be inspected coincident with core barrel removal.

4.3 PRIMARY SYSTEM SURVEILLANCE (Contd)

Item 2.2 has been modified to exclude penetrations of one inch or less pipe size since it is not intended to cover small instrument connections under this code.

Item 2.3 has been modified to delete the surface examination since the pressurizer heater penetration welds are not accessible to a surface examination.

Item 2.4 has been deleted since visual examination of the heater penetrations is covered under Item 2.3.

Item 2.5 has been deleted since no bolting 2-inch and over is used.

Item 2.7 - The reference to volumetric inspection has been deleted since these are partial penetration welds which cannot be tested by volumetric means.

Item 3.1 - The reference to longitudinal and shell welds has been deleted since neither exists in the design.

Item 3.3 - See discussion of Item 1.7.

Item 3.4 has been deleted since no bolting 2-inch and over is used.

Item 4.1 - Reference to vessel and valve safe ends has been deleted since the plant has no valves and no vessel safe ends. The reference to surface examination has been deleted since the pump to primary pipe safe ends are Inconel rather than stainless steel which is the material of concern.

Items 5.1 and 5.2 - These items have been deleted since the pump insulation does not permit visual examination and the pump material and design do not permit a meaningful volumetric examination by either method.

Item 6.1 has been deleted since seam welded valves have not been used.

Item 6.4 has been deleted since no bolting 2-inches and over is used.

Table 4.3.3 is consistent with the surveillance program as presented in the FSAR.[2] However, the withdrawal schedule has been modified to reflect the slightly different wall fluence values resulting from removal of the thermal shield.

References

(1) FSAR, Section 4.5.6.

(2) FSAR, Section 4.5.3.

Table 4.3.1

Primary Coolant System Surveillance

Item Number	Examination Category per Table IS251	Areas To Be Examined	Method
		Reactor Vessel and Closure Head	
1.1	A	Longitudinal and circumferential shell welds in core region.	Volumetric
1.2	B	Longitudinal and circumferential welds in shell (other than those of Categories A and C).	Volumetric
1.3	C	Vessel-to-flange and head-to-flange circumferential welds.	Volumetric
1.4	D	Primary nozzle-to-vessel welds and nozzle-to-vessel inside radiused section.	Volumetric
1.5	E-1	Not applicable.	
1.6	E-2	Vessel penetrations, including control rod drive penetrations and control rod housing pressure boundary welds.	Visual
1.7	J	Primary nozzles to transition piece.	Volumetric
1.8	G-1	Closure studs and nuts.	Volumetric and Visual or Surface
1.9	G-1	Ligaments between threaded stud holes.	Volumetric
1.10	G-1	Closure washers.	Visual
1.11	G-2	Not applicable.	
1.12	H	Not applicable.	
1.13	I-1	Closure head cladding.	Visual and Surface or Volumetric
1.14	I-1	Vessel cladding.	Visual
1.15	N*	Interior surfaces and internals.	Visual

*Core support barrel bottom snubbers and drop prevention supports will be inspected coincident with core barrel removal.

Table 4.3.1 (Contd)

Primary Coolant System Surveillance (Contd)

Item Number	Examination Category per Table IS251	Areas To Be Examined	Method
		Pressurizer	
2.1	B	Longitudinal and circumferential welds.	Visual and Volumetric
2.2	D	Nozzle-to-vessel welds (greater than "1").	Visual and Volumetric
2.3	E-1	Heater connections.	Visual
2.4	E-2	Not applicable.	
2.5	G-1	Not applicable.	
2.6	G-2	Pressure retaining bolting.	Visual
2.7	H	Integrally welded vessel supports.	Visual
2.8	I-2	Vessel cladding.	Visual
		Steam Generators	
3.1	B	Circumferential welds, including tube-sheet-to-head welds on the primary side.	Visual and Volumetric
3.2	D	Primary nozzle-to-vessel head welds and nozzle-to-head inside radiused section.	Visual and Volumetric
3.3	J	Primary nozzle to transition piece.	Visual and Volumetric
3.4	G-1	Not applicable.	
3.5	G-2	Pressure retaining bolting.	Visual
3.6	H	Integrally welded vessel supports	Visual and Volumetric
3.7	I-2	Vessel cladding.	Visual

Table 4.3.1 (Contd)

Primary Coolant System Surveillance (Contd)

Item Number	Examination Category per Table IS251	Areas To Be Examined	Method
		Piping Pressure Boundary	
4.1	F	Pump safe-ends to primary pipe to primary pipe welds and safe-ends in branch piping welds.	Visual and Volumetric
4.2	J	Circumferential and longitudinal pipe welds.	Visual and Volumetric
4.3	G-1	Not applicable.	
4.4	G-2	Pressure-retaining bolting.	Visual
4.5	K-1	Integrally welded supports.	*Visual and Volumetric
4.6	K-2	Piping support and hanger.	Visual
		Pump Pressure Boundary	
5.1	L-1	Not applicable.	
5.2	L-2	Not applicable.	
5.3	F	Nozzle-to-safe end welds.	Visual and Volumetric
5.4	G-1	Pressure-retaining bolting.	Visual and Volumetric
5.5	G-2	Pressure-retaining bolting.	Visual
5.6	K-1	Integrally welded supports.	*Visual and Volumetric
5.7	K-2	Supports and hangers	Visual
		Valve Pressure Boundary	
6.1	M-1	Not applicable.	

*Volumetric examination not applicable to partial penetration welds.

Table 4.3.1 (Contd)

Primary Coolant System Surveillance (Contd)

Item Number	Examination Category per Table IS251	Areas To Be Examined	Method
		Valve Pressure Boundary (Contd)	
6.2	M-2	Valve bodies.	Visual
6.3	F	Valve-to-safe end welds.	Visual and Volumetric
6.4	G-1	Not applicable.	
6.5	G-2	Pressure-retaining bolting.	Visual and Volumetric
6.6	K-1	Integrally welded supports.	*Visual and Volumetric
6.7	K-2	Supports and Hangers.	Visual

*Volumetric examination not applicable to partial penetration welds.

Table 4.3.2

Miscellaneous Surveillance Items

Equipment	Method	Frequency
1. Regenerative Heat Exchanger		
a. Primary Side Shell to Tube Sheet Welds	Volumetric	5-Year Maximum Interval (100%)
b. Primary Head	Volumetric	5-Year Maximum Interval (100%)
2. Primary Coolant Pump Flywheels	Volumetric	100% Upper Flywheel Each Refueling

Table 4.3.3

Reactor Vessel Surveillance Coupon Removal Schedule

Refueling Schedule	Capsule Removed	Estimated Target Fluence N/cm^2	Integrated Power
2	A-60	1.7×10^{19}	1.73×10^{6}
5	A-240	3.5×10^{19}	3.97×10^{6}
5	W-290	3.7×10^{18}	3.97×10^{6}
5	T-330	-	3.97×10^{6}
11	W-110	9.3×10^{18}	8.44×10^{6}
20	W-100	1.7×10^{19}	1.51×10^{7}
20	T-150	-	1.51×10^{7}
25	W-280	2.1×10^{19}	1.89×10^{7}
35	W-260	3.0×10^{19}	2.63×10^{7}
40	W-80	3.4×10^{19}	3.0×10^{7}

4.4 PRIMARY COOLANT SYSTEM INTEGRITY TESTING

Applicability

Applies to test requirements for primary coolant system integrity.

Objective

To specify test for primary coolant system integrity after the system is closed following normal opening, modification or repair.

Specifications

a. Whenever the primary coolant system is closed after it has been opened, the system shall be leak tested at not less than 2135 psig prior to the reactor being made critical.

b. Whenever modifications or repairs are made in the primary coolant system that involve new strength welds on components greater than 2-inch diameter, the new welds shall receive both a surface and 100% volumetric examination and shall meet all applicable code requirements.

c. Whenever modifications or repairs are made in the primary coolant system that involve new strength welds on components 2-inch diameter or smaller, the new welds shall receive a surface examination.

Basis

For normal opening, the integrity of the primary coolant system, in terms of strength, is unchanged. If the system does not leak at 2135 psig (operating pressure + 50 psi; $\pm$ 50 psi is normal system pressure fluctuation),[1] it will be leak tight during normal operation. If the pressure goes above 2135 psig, the worst consequence is a leak.

For repairs on components greater than 2-inch diameter, the thorough nondestructive testing gives a very high degree of confidence in the integrity of the primary coolant system and will detect any significant defects in and near the new welds.

Repairs on components 2-inch diameter or smaller are relatively minor in comparison and the surface examination assures a similar standard of integrity. In all cases, the leak test will insure leak tightness during normal operation.

References

(1) FSAR, Section 4.2.

4.5 CONTAINMENT TESTS

Applicability

Applies to containment leakage and structural integrity.

Objective

To verify that potential leakage from the containment and the prestressing tendon loads are maintained within specified values.

Specifications

4.5.1 Integrated Leakage Rate Tests

a. Test

(1) Integrated leak rate tests shall be performed prior to initial plant operations at containment design pressure (P_p) of 55 psig and a test pressure (P_t) of at least 28 psig to establish the respective measured leak rates, L_{pm} and L_{tm}. A minimum test temperature of 50°F will be utilized. The maximum test temperature will be 100°F.

(2) Subsequent leak rate tests shall be performed at the test pressure of about 28 psig. The tests shall be performed without any leak detection surveys or leak repairs immediately prior to or during the test, except as noted below.

(3) Major leak repairs, if necessary to permit the integrated leak rate test, shall be preceded by local leakage measurements. The local leakage differences, as a result of repair, shall be corrected to P_t and added to the final integrated leak rate test result to determine the subsequent retest schedule.

(4) All systems which, under accident conditions, become an extension of the containment shall be vented to the containment atmosphere during integrated leak rate tests. Closure of containment isolation valves is to be accomplished by the normal mode of actuation.

(5) The test duration shall not be less than 24 hours unless test experiences of at least two prior tests provide evidence of the adequacy of shorter test duration. Test accuracy shall be verified by supplementary means, such as measuring the quantity of air required to return to the starting point, or by imposing a known leak rate to demonstrate validity of measurements.

4.5 CONTAINMENT TESTS (Contd)

b. Acceptance Criteria

(1) The maximum allowable leakage rate under DBA conditions, L_a, shall not exceed 0.10 weight percent per 24 hours.

(2) The allowable operational leakage rate, L_{to}, which shall be met prior to resumption of power operation following a test (either as measured or following repairs and retest) shall not exceed $0.8L_t$ (for a two-year test interval) or $0.7L_t$ (for a three-year test interval) where L_t is defined by 4.5.1.b(3).

(3) The allowable leakage rate, L_t, at the reduced test pressure, shall not exceed the lesser of $L_p(L_{tm}/L_{pm})$ or $L_p(P_t/P_p)^{1/2}$.

L_p is defined by $L_p = L_a(R_pT_p/R_aT_a)^{1/2}$ and corresponds to the maximum allowable leakage that would be measured if the containment was pressurized with air to the accident pressure, P_p. The subscript m refers to values of the leakage measured during preoperational tests. The subscripts p and t refer to tests at accident pressure and reduced pressure, respectively. Subscript a refers to accident conditions at accident pressure.

c. Corrective Action for Retests

(1) If repairs are necessary to meet the acceptance criterion, the integrated leak rate test need not be repeated provided local leakage rate measurements are made and the leakage rate difference achieved by repairs reduces the overall measured integrated leak rate to a value not in excess of the allowable operational leakage rate, L_{to}.

(2) The reduction in leakage effected by the repair of isolation valves shall be included in the integrated leak rate test results.

d. Frequency

(1) Integrated leak rate tests shall be performed as follows, except as noted in 4.5.1.4.b and 4.5.1.4.c.

(a) At the first refueling shutdown.

(b) Every third refueling shutdown following a previous test.

(2) The intervals between integrated leak rate tests shall not exceed 24 months from the date of initial criticality, and 45 months between successive tests thereafter.

4.5 CONTAINMENT TESTS (Contd)

(3) In the event it is necessary to make repairs during any integrated leak rate test in order to meet the acceptance criteria specified, the allowable operational leak rate (L_{to}) applicable to the next scheduled integrated leak rate test shall be reduced by the leakage measured before repairs minus L_t.

e. Report of Test Results

Each integrated leak rate test will be the subject of a summary technical report, which will include summaries of local leak detection tests and leak test of the recirculation heat removal systems.

4.5.2 Local Leak Detection Tests

a. Test

(1) Local leak rate tests shall be performed at a pressure of not less than 55 psig.

(2) Acceptable methods of testing are halogen gas detection, soap bubble, pressure decay, or equivalent.

(3) The local leak rate shall be measured for each of the following components:

(a) Containment penetrations that employ resilient seal gaskets, sealant compounds, or bellows.

(b) Air lock and equipment door seals.

(c) Fuel transfer tube.

(d) Isolation valves on the testable fluid systems' lines penetrating the containment.

(e) Other containment components, which require leak repair in order to meet the acceptance criterion for any integrated leak rate test.

b. Acceptance Criterion

The total leakage from all penetrations and isolation valves shall not exceed $0.45L_p$.

c. Corrective Action

(1) If at any time it is determined that $0.45L_p$ is exceeded, repairs shall be initiated immediately.

(2) If repairs are not completed and conformance to the acceptance criterion of 4.5.2.b is not demonstrated within 48 hours, the reactor shall be shut down and depressurized until repairs are effected and the local leakage meets this acceptance criterion.

d. Test Frequency

(1) Individual penetrations and containment isolation valves shall be leak rate tested at a frequency of at least every six months prior to the first postoperational integrated leak rate test and at a frequency of at least every refueling thereafter, except as specified in (a) and (b) below:

(a) The containment equipment hatch and the fuel transfer tube shall be tested at each refueling shutdown or after each time used, if that be sooner.

(b) The personnel air lock seals shall be tested at six-month intervals, except when the air locks are not opened during the interval. In that case, the test is to be performed after each opening, except that no test interval is to exceed twelve months.

(2) Each three months the isolation valves must be stroked to the position required to fulfill their safety function unless it is established that such operation is not practical during plant operation. The latter valves shall be full-stroked during each cold shutdown.

4.5.3 Recirculation Heat Removal Systems

a. Test

(1) The portion of the shutdown cooling system that is outside the containment shall be tested either by use in normal operation or hydrostatically tested at 255 psig at the interval specified in 4.5.3.d.

(2) Piping from valves CV-3029 and CV-3030 to the discharge of the safety injection pumps and containment spray pumps shall be hydrostatically tested at no less than 100 psig at the interval specified in 4.5.3.d.

4.5 CONTAINMENT TESTS (Contd)

(3) Visual inspection shall be made for excessive leakage from components of the system. Any significant leakage shall be measured by collection and weighing or by another equivalent method.

b. Acceptance Criterion

The maximum allowable leakage from the recirculation heat removal systems' components (which include valve stems, flanges and pump seals) shall not exceed one-half gallon per minute under the normal hydrostatic head from the SIRW tank (approximately 44 psig).

c. Corrective Action

Repairs shall be made as required to maintain leakage within the acceptance criterion of 4.5.3.b.

d. Test Frequency

Tests of the recirculation heat removal system shall be conducted at intervals not to exceed twelve months.

4.5.4 Surveillance for Prestressing System, End Anchorage Concrete, Liner Plate and Penetrations

a. The nine surveillance tendons shall be periodically inspected for symptoms of material deterioration or force reduction. The surveillance tendons consist of three horizontal tendons, one in each of three 120° sectors of the containment; three vertical tendons located at approximately 120° apart; and three dome tendons located approximately 120° apart.

b. The inspection intervals, measured from the date of the initial structural test, shall be as follows:

(1) One year.

(2) Three years.

(3) Every five years thereafter.

c. Each surveillance tendon shall be inspected for the following:

(1) The number of broken wires.

(2) The surveillance tendon force at the end anchor.

(3) The corrosion and other physical conditions for the tendons, including bearing plates and end anchor assemblies.

4.5 CONTAINMENT TESTS (Contd)

The inspection procedures include removal, inspection and testing of specimens from not less than one nor more than five wires from each of the nine tendons. The grease will be sampled and tested.

d. Comparisons shall be made between the quality control records and each of the surveillance inspection records for each of the surveillance tendons. Acceptance criteria for paragraph (c) is as follows:
   (1) No additional broken wires since last inspection.
   (2) The force-time trend line for each tendon as extrapolated shall not intersect either the upper or lower bounds of the predicted design band.
   (3) No unexpected change in corrosion conditions or grease properties.

e. If the criteria stated in paragraph (d) are not met, an investigation shall be undertaken to determine and correct the cause for the changes and a report made to the AEC as specified in Section 6.6.

4.5.5 End Anchorage Concrete Surveillance

a. Specific locations for surveillance will be chosen from the combined information from the design calculations; the as-built end anchorage concrete and prestressing records; observations of the end anchorage concrete during and after prestressing; and the results of strain and deformation measurements made during prestressing and the initial structural test.

b. The inspection intervals will be approximately one-half year and one year after the initial structural test and shall be chosen such that the inspection occurs during the warmest and coldest part of the year following the initial structural test.

c. The inspections made shall include:
   (1) Visual inspection of the end anchorage concrete exterior surfaces.
   (2) A determination of the temperatures of the liner plate area or containment interior surface in locations nearest to the end anchorage concrete under surveillance.
   (3) Measurement of concrete temperatures at specific end anchorage concrete surfaces being inspected.
   (4) The mapping of the predominant visible concrete crack patterns.

4.5 CONTAINMENT TESTS (Contd)

(5) The measurement of the crack widths, by use of optical comparators or wire feeler gauges.

(6) The measurement of movements, if any, by use of demountable mechanical extensometers.

d. The measurements and observations shall be compared with those to which prestressed structures have been subjected in normal and abnormal load conditions and with those of preceding measurements and observations at the same location on the reactor building.

e. The acceptance criteria shall be as follows:

If the inspections determine that the conditions are favorable in comparison with experience and predictions, the close inspections will be terminated by the last of the inspections stated in the schedule and a report will be prepared which documents the findings and recommends the schedule for future inspections, if any. If the inspections detect symptoms of greater than normal cracking or movements, an immediate investigation will be made to determine the cause.

4.5.6 Liner Plate

a. The liner plate will be examined before the initial pressure test to determine the following:

(1) Locate areas which have inward deformations. The magnitude of the inward deformations will be measured and recorded. The areas will be permanently marked for future reference. The inward deformations will be measured between the angle stiffeners which are on 15-inch centers. The measurements will be accurate to $\pm$ .01 inch.

(2) Try to locate areas having strain concentrations by visual examination paying particular attention to the condition of the liner surface. Record the location of any areas having strain concentrations.

b. Shortly after the initial pressure test and at one year after initial start-up, reexamine the areas located in section (a). Measure and record inward deformations. Record observations pertaining to strain concentrations.

4.5 CONTAINMENT TESTS (Contd)

c. If the difference in the measured inward deformations exceeds 0.25 inch (for a particular location) and/or changes in strain concentration exist, then an investigation will be made. The investigation will determine the cause and any necessary corrective action.

d. The surveillance program will only be continued beyond the one year after initial start-up inspection if some corrective action was needed. The frequency of inspection for a continued surveillance program will be determined shortly after the "one year after initial start-up inspection."

e. In addition to the preceding requirements, temperature readings will be obtained at the locations where inward deformations were measured. Temperature measurements will also be obtained on the opposite side of the containment building wall.

4.5.7 Penetrations

a. The penetration assemblies will be visually inspected before the initial pressure test, shortly after the initial pressure test and one year after initial start-up.

b. The inspector will place particular emphasis on the nozzle-to-penetration reinforcing plate weld and all other closure welds.

c. The inspector will concentrate on finding any indications which might affect the leak tightness or structural integrity such as weld cracks, pinholes, flaws, etc.

d. If any indications are found, further examination using nondestructive testing techniques will be used to verify or disprove the visual examination. All indications will be documented stating type of indication, location, NDT used for verification and final results.

e. For any verified indication, an examination will be conducted to determine the cause and any necessary corrective action.

f. The surveillance program will only be continued beyond the one year after initial start-up inspection if some corrective action was needed. The frequency of inspection for a continued surveillance program will be determined shortly after the one year after initial start-up inspection.

4.5 CONTAINMENT TESTS (Contd)

Basis

The containment is designed for an accident pressure of 55 psig.(1) While the reactor is operating, the internal environment of the containment will be air at approximately atmospheric pressure and a temperature of about $104^{\circ}F$. With these initial conditions, the temperature of the steam-air mixture at the peak accident pressure of 55 psig is $283^{\circ}F$.

Prior to initial operation, the containment will be strength-tested at 63 psig and then will be leak rate tested. The design objective of this preoperational leak rate test has been established as 0.1% by weight per 24 hours at 55 psig. This leakage rate is consistent with the construction of the containment,(2) which is equipped with independent leak-testable penetrations and contains channels over all unaccessible containment liner welds, which are independently leak-tested during construction.

Accident analyses have been performed on the basis of a leakage rate of 0.1% by weight per 24 hours for the first 24 hours during an accident. With this leakage rate and with a reactor power level of 2200 MWt, the potential public exposure would be below 10 CFR 100 guideline values in the event of the design basis loss-of-coolant accident.(3)

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

This normal manner is a coincident two-of-four high radiation or two-of-four high containment pressure signals which will close all containment isolation valves not required for engineered safety features except the component cooling lines' valves which are closed by SIS. The control system is designed on a two-channel (right and left) concept with redundancy and physical separation. Each channel is capable of initiating containment isolation.(4)

4.5 CONTAINMENT TESTS (Contd)

The test pressure of 28 psig for the periodic integrated leak rate test is sufficiently high to provide an accurate measurement of the leakage rate and it duplicates the preoperational leak rate test at 28 psig. The specification provides relationships for relating in a conservative manner the measured leakage of air at 28 psig to the potential leakage of a steam-air mixture at 55 psig and 283°F. The specification also allows for possible deterioration of the leakage rate between tests, by requiring that only 70% or 80% of the allowable leakage rates actually be measured. The basis for these deterioration allowances is arbitrary judgments, which are believed to be conservative and which will be confirmed or denied by periodic testing. If indicated to be necessary, the deterioration allowances will be altered based on experience.

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

The frequency of the periodic integrated leak rate test is keyed to the refueling schedule for the reactor, because these tests can best be performed during refueling shutdowns. The initial core loading is designed for approximately 18 months of power operation; thus, the first refueling will occur approximately 24 months after initial criticality. Subsequent refueling shutdowns are scheduled at approximately 12-month intervals. The specified frequency of periodic integrated leak rate tests is based on three major considerations. First is the low probability of leaks in the liner, because of (a) the test of the leak tightness of the welds during erection; (b) conformance of the complete containment to a low leak rate at 55 psig during preoperational testing which is consistent with 0.1% leakage at design basis accident (DBA) conditions; and (c) absence of any significant stresses in the liner during reactor operation. Second is the more frequent testing, at the full accident pressure, of those portions of the containment envelope that are most likely to develop leaks during reactor operation (penetrations and isolation valves) and the low value ($0.45L_p$) of the total leakage that is specified as acceptable from penetrations and isolation valves. Third is the tendon stress surveillance program, which provides assurance that

an important part of the structural integrity of the containment is maintained.

The basis for specification of a total leakage rate of $0.45L_p$ from penetrations and isolation valves is that only one-half of the allowable integrated leak rate should be from those sources, in order to provide assurance that the integrated leak rate would remain within the specified limits during the intervals between integrated leak rate tests. The allowable value of $0.50L_p$ has then been reduced 10% to allow for possible deterioration in the intervals (normally six months or 18 months) between tests. The limiting leakage rates from the shutdown cooling system are judgment values based primarily on assuring that the components could operate without mechanical failure for a period on the order of 200 days after a DBA. The test pressure (270 psig) achieved either by normal system operation or by hydrostatically testing gives an adequate margin over the highest pressure within the system after a DBA. Similarly, the hydrostatic test pressure for the return lines from the containment to the shutdown cooling system (100 psig) gives an adequate margin over the highest pressure within the lines after a DBA.[5]

A shutdown cooling system leakage of 1/2 gpm will limit off-site exposures due to leakage to insignificant levels relative to those calculated for leakage directly from the containment in the DBA. The engineered safeguards room ventilation system is equipped with isolation valves which close upon a high-radiation signal from a local radiation detector. These monitors shall be set at 2.2×10^5 cpm, which is well below the expected level, following a loss-of-coolant accident (LOCA), even without clad failure. The 1/2-gpm leak rate is sufficiently high to permit prompt detection and to allow for reasonable leakage through the pump seals and valve packings, and yet small enough to be readily handled by the sumps and radiate system. Leakage to the engineered safeguards room sumps will be returned to the containment clean water receiver following an LOCA, via the equipment drain tank and pumps. Additional makeup

4.5 CONTAINMENT TESTS (Contd)

water to the containment sump inventory can be readily accommodated via the charging pumps from either the SIRW tank or the concentrated boric acid storage tanks.

In case of failure to meet the acceptance criteria for leakage from the shutdown cooling system or the penetrations, it may be possible to effect repairs within a short time. If so, it is considered unnecessary and unjustified to shut down the reactor. The times allowed for repairs are consistent with the times developed for other engineered safety feature components.

A reduction in prestressing force and changes in physical conditions are expected for the prestressing system. Allowances have been made in the reactor building design for the reduction and changes. The inspection results for each tendon shall be recorded on the forms provided for that purpose and comparison will be made with the previous test results and the initial quality control records.

Force-time trend lines will also be established and maintained for each of the surveillance tendons.

If the force-time trend line, as extrapolated, intersects the lower bound curve of the banded force-time graph for one or more surveillance tendons, and before the next scheduled surveillance inspection, an investigation shall be made to determine whether the rate of force reduction is indeed occurring for other tendons. If the rate of reduction is confirmed, the investigation shall be extended so as to identify the cause of the rate of force reduction. The extension of the investigation shall determine the needed changes in the surveillance inspection schedule and the criteria and initial planning for corrective action.

If the force-time trend lines of the surveillance tendons at any time exceed the upper bound curve of the band on the force-time graph, an investigation shall be made to determine the cause.

If the comparison of corrosion conditions, including chemical tests of the corrosion protection material, indicate a larger than expected change in the conditions from the time of installation or last surveillance inspection, an investigation shall be made to detect and correct the causes.[6, 7]

4.5 CONTAINMENT TESTS (Contd)

The prestressing system is a necessary strength element of the plant safeguards and it is considered desirable to confirm that the allowances are not being exceeded. The technique chosen for surveillance is based upon the rate of change of force and physical conditions so that the surveillance can either confirm that the allowances are sufficient, or require maintenance before minimum levels of force or physical conditions are reached.

The end anchorage concrete is needed to maintain the prestressing forces. The design investigations have concluded that the design is adequate. The prestressing sequence has shown that the end anchorage concrete can withstand loads in excess of those which result when the tendons are anchored. At the time of initial pressure testing, the containment building will have been subjected to temperature gradients equivalent to those for normal operating conditions while the prestressing tendon loads are at their maximum.

However, both concrete creep and prestressing losses will increase with the greatest rapidity in the year after the initial pressure test and result in a redistribution of the stresses and a reduction in end anchor force. Because of the importance of the containment and the fact that the design is new, it is considered prudent to continue the surveillance for this initial period.

Since the liner plate and the penetration assemblies are of a relatively new design, a surveillance program will be used to monitor the behavior of the liner plate and the penetration assemblies. The surveillance program will provide further proof as to the integrity of the system. The liner plate examination will include taking inward deformation measurements and making observations of strain concentrations. The penetration assembly examination will be concerned with locating any indications which might lead to a partial loss-of-structural or leak-tight integrity. Inspection will be performed before and after the pressure test and at the first refueling which is approximately two years after the pressure test. Also, if the reactor is shut down for approximately

4.5 CONTAINMENT TESTS (Contd)

two weeks (in extreme weather) during this two-year period, a visual inspection shall be made for stress indications. After this time, if no unexpected behavior of the liner plate or penetration assemblies is observed, then the surveillance program will be terminated since the system is not expected to experience more severe conditions under future normal operation. If unexpected behavior of the system is observed, the surveillance program will be extended at that time.

References

(1) FSAR, Section 5.1.2.

(2) FSAR, Section 5.1.8.

(3) FSAR, Section 14.22.

(4) FSAR, Section 8.5.4.

(5) FSAR, Section 6.2.3.

(6) FSAR, Section 5.1.8.4.

(7) FSAR, Amendment No. 14, Question 5.37.

4.6 SAFETY INJECTION AND CONTAINMENT SPRAY SYSTEMS TESTS

Applicability

Applies to the safety injection system, the containment spray system, chemical injection system and the containment cooling system tests.

Objective

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

Specifications

4.6.1 Safety Injection System

a. System tests shall be performed at each reactor refueling interval. A test safety injection signal will be applied to initiate operation of the system. The safety injection and shutdown cooling system pump motors may be de-energized for this test.

b. The system 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 (ie, the appropriate pump breakers shall have opened and closed, and all valves shall have completed their travel).

4.6.2 Containment Spray System

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

b. At least every five years the spray nozzles shall be verified to be open.

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

4.6.3 Pumps

a. The safety injection pumps, shutdown cooling pumps, and containment spray pumps shall be started at intervals not to exceed three months. Alternate manual starting between control room console and the C-33 panel shall be practiced in the test program.

4.6 SAFETY INJECTION AND CONTAINMENT SPRAY SYSTEMS TESTS (Contd)

b. Acceptable levels of performance shall be that the pumps start, reach their rated shutoff heads at minimum recirculation flow, and operate for at least fifteen minutes.

4.6.4 Valves

a. The SIRW storage tank outlet valves and containment sump isolation valves shall be exercised during the pump tests.

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

c. Actuation of valves shall be alternated between control room panels and C-33 panel.

4.6.5 Containment Air Cooling System

a. Emergency mode automatic valve and fan operation will be checked for operability during each refueling shutdown.

b. Each fan and valve required to function during accident conditions will be exercised at intervals not to exceed three months.

Basis

The safety injection system and the containment spray system are principal plant safety features that are normally inoperative during reactor operation.

Complete systems tests cannot be performed when the reactor is operating because a safety injection signal causes 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. A test signal is applied to initiate automatic action and verification made that the components receive the safety injection in the proper sequence. The test demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry.(1, 2)

4.6 SAFETY INJECTION AND CONTAINMENT SPRAY SYSTEMS TESTS (Contd)

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 addition, the active components (pumps and valves) are to be tested every three months to check the operation of the starting circuits and to verify that the pumps are in satisfactory running order. The test interval of three months is based on the judgment that more frequent testing would not significantly increase the reliability (ie, the probability that the component would operate when required), yet more frequent test would result in increased wear over a long period of time. Verification that the spray piping and nozzles are open will be made initially by a smoke test or other suitably sensitive method, and at least every five years thereafter. Since the material is all stainless steel, normally in a dry condition, and with no plugging mechanism available, the retest every five years is considered to be more than adequate.

Other systems that are also important to the emergency cooling function are the SI tanks, the component cooling system, the service water system and the containment air coolers. The SI tanks are a passive safety feature. In accordance with the specifications, the water volume and pressure in the SI tanks are checked periodically. The other systems mentioned operate when the reactor is in operation and by these means are continuously monitored for satisfactory performance.

References

(1) FSAR, Section 6.1.3.

(2) FSAR, Section 6.2.3.

4.7 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.

Specifications

4.7.1 Diesel Generators

a. Each diesel generator shall be manually started each month and demonstrated to be ready for loading within 10 seconds. The signal initiated to start the diesel shall be varied from one test to another to verify all starting circuits are operable. The generation shall be synchronized from the control room, paralled and loaded to the nameplate rating.

b. A test shall be conducted during each refueling outage to demonstrate the overall automatic operation of the emergency power system. The test shall be initiated by a simulated simultaneous loss of normal and standby power sources and a simulated SIS signal. Proper operations shall be verified by bus load shedding and automatic starting of selected motors and equipment to establish that restoration with emergency power has been accomplished within 30 seconds.

c. Each diesel generator shall be given a thorough inspection at least annually following the manufacturer's recommendations for this class of standby service. The above tests will be considered satisfactory if all applicable equipment operates as designed.

d. Diesel generator electric loads shall not be increased beyond the continuous rating of 2500 kW.

e. The fuel transfer pumps shall be verified to be operable each month.

4.7.2 Station Batteries

a. Every month, the voltage of each cell (to the nearest 0.01 volt), the specific gravity and the temperature of a pilot cell in each battery shall be measured and recorded.

4.7 EMERGENCY POWER SYSTEM PERIODIC TESTS (Contd)

b. Every three months, 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.

4.7.3 Emergency Lighting

The correct functioning of the emergency lighting system shall be verified at least once each year.

Basis

The emergency power system provides power requirements for the engineered safety features in the event of a DBA. Each of the two diesel generators is capable of supplying minimum required safeguards equipment from independent buses.[1, 2] This redundancy is a factor in establishing testing intervals. The monthly tests specified above will demonstrate operability and load capacity of the diesel generator. The fuel supply and various controls are continuously monitored and alarmed for abnormal conditions. Starting on complete loss of off-site power will be verified by simulated loss-of-power tests at approximately yearly intervals (during refueling shutdowns). Considering system redundancy, the specified testing intervals for the station batteries should be adequate to detect and correct any malfunction before it can result in system malfunction. 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.

References

(1) FSAR, Section 8.4.1.

(2) FSAR, Section 8.5.2.2.

4.8 MAIN STEAM STOP VALVES

Applicability

Applies to periodic testing of the main steam stop valves.

Objective

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

Specifications

The operation of the main steam stop valves shall be tested during each refueling shutdown to demonstrate a closure time of five seconds or less under no-flow conditions.

Basis

The main steam stop valves serve to limit an excessive primary coolant system cooldown rate and resultant reactivity insertion following a main steam break incident. Their ability to close upon signal should be verified at each scheduled refueling shutdown.

References

FSAR, Sections 7.2.3.8 and 14.14.

4.9 AUXILIARY FEED-WATER SYSTEM

Applicability

Applies to periodic testing requirements of the turbine-driven and motor-driven auxiliary feed-water pumps.

Objective

To verify the operability of the auxiliary feed-water system and its ability to respond properly when required.

Specifications

a. The operability of the motor-driven auxiliary feed-water pump will be confirmed at least every three (3) months.

b. The operability of the steam turbine-driven auxiliary feed-water pump will be confirmed at least every three (3) months.

c. The operability of the auxiliary feed-water pumps' discharge valves (Cv 0736 and Cv 0737) shall be confirmed at least every three (3) months.

Basis

The testing each three months of the auxiliary feed-water pumps will verify their operability by recirculating water to the condensate storage tank and simultaneously partially opening, one at a time, the discharge valve (Cv 0736 and Cv 0737) to confirm a flow path to the steam generators.

Proper functioning of the steam turbine admission valve and the feed-water pumps' start will demonstrate the integrity of the steam-driven pump. Verification of correct operation will be made both from instrumentation within the main control room and direct visual observation of the pumps.

References

FSAR, Section 9.7

4.10 REACTIVITY ANOMALIES

Applicability

Applies to potential reactivity anomalies.

Objective

To require evaluation of reactivity anomalies within the reactor.

Specifications

Following a normalization of the computed boron concentration as a function of burnup, the actual boron concentration of the primary coolant shall be periodically compared with the predicted value. If the difference between the observed and predicted steady-state concentrations reaches the equivalent of 1% in reactivity, the Atomic Energy Commission shall be notified within 24 hours and an evaluation as to the cause of the discrepancy shall be made and reported to the Atomic Energy Commission within 30 days.

Basis

To eliminate possible errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burnup and the boron concentration, necessary to maintain adequate control characteristics, must be adjusted (normalized) to accurately reflect actual core conditions. When full power is reached initially, and with the control rod groups in the desired positions, the boron concentration is measured and the predicted curve is adjusted to this point. As power operation proceeds, the measured boron concentration is compared with the predicted concentration and the slope of the curve relating burnup and reactivity is compared with that predicted. This process of normalization should be completed after about 10% of the total core burnup. Thereafter, actual boron concentration can be compared with prediction and the reactivity status of the core can be continuously evaluated. Any reactivity anomaly greater than 1% would be unexpected, and its occurrence would be thoroughly investigated and evaluated. The methods employed in calculating the reactivity of the core vs burnup and the reactivity worth of boron vs burnup are given in the FSAR.

The value of 1% is considered a safe limit since a shutdown margin of at least 2% with the most reactive rod in the fully withdrawn position is always maintained.[1]

References

(1) FSAR, Section 3.3.2

4.11 ENVIRONMENTAL RADIATION SURVEY

Applicability

Applies to routine testing of plant environs.

Objective

To establish a sampling schedule which will assure recognition of changes in radioactivity in the environs.

4.11.1 **Specifications**

Environmental samples shall be taken according to the following schedule:

Table 4.11.1

Specific Samples and Collection Frequency

Sample Class	Collection Frequency	Amount To Be Collected (Operational)
Air	Weekly	12
Lake Water	Monthly	2
Well Water	Monthly	3
Milk	Monthly	4
Organic	Monthly in Season	Crops and Fish As Desired
Film or TLD	Monthly in Season	21
Lake Bottom Sediment	Twice per Season	4

4.11.2 The sensitivities listed below and in Table 4.11.2 shall be used for the samples listed in Table 4.11.1.

Air - When a gross beta count reveals radioactivity levels in excess of 1×10^{-13} μCi/ml, analyses for specific isotopes as listed will be performed with the exception of I-131. An I-131 analysis will be performed on all air samples.

Water - When a gross beta count reveals radioactivity in excess of 1×10^{-8} μCi/ml, all other specific analyses listed will be performed with the following exception: Tritium analyses will be performed on all Lake Michigan water samples.

Organic - When a gross beta count reveals radioactivity in excess of 5×10^{-8} μCi/ml, all other listed analyses will be performed.

Milk - Listed analyses will be performed on all samples.

4.11 ENVIRONMENTAL RADIATION SURVEY (Contd)

Basis

The environmental surveillance program for Palisades is designed to meet the following objectives:

1. Measurement of radiation levels in the sampled media is done in such a manner to assure compliance with 10 CFR 20.
2. Survey design is such that releases of plant origin can be differentiated from natural or other sources of environmental radiation. This is accomplished in two ways. First of all, the commonly called "reference area approach to environmental surveillance" is used. This makes use of a calculation that shows whether or not a statistical difference exists between the levels of radioactivity detected near the site and those detected remote from the site. Secondly, specific isotopic analyses are performed. In this manner, concentrations of specific isotopes in the sampled environmental media can be related to known plant releases of the same nuclide.
3. Dose estimates to man are made if significant increases in radiation levels have been found to occur as a result of (2) above. This is meaningfully done by (a) requiring detection sensitivity to be far below any effective maximum permissible concentration, and (b) planning the survey with sample collection designed to estimate dose.

The survey consists of 12 stations. Nine are near and three are remote from the site. In addition, nine extra film-TLD stations are located at the site boundary and lake water, biota and bottom sediment are further collected at the site. In this way, inhalation, ingestion and direct dose can be estimated and because of the Lake Michigan sampling, food chain parameters can be determined. Program description including sample collection frequency and detection sensitivity is outlined in Specification 4.11.

4.11 ENVIRONMENTAL RADIATION SURVEY (Contd)

Design and Methods of the Environmental Survey

Meeting the objective of providing public assurance that the facility's contribution to the naturally existing radioactivity in the environment is negligible requires analyses to be performed such that sensitivities are far below those resulting in maximum permissible dose (MPD) to man. In other words, levels of radioactivity in water, air or food to be detected shall be far below those resulting in MPD to man. Levels of radioactivity in water, air or food to be detected shall be a small fraction of any effective maximum permissible concentration (MPC) allowed in such a sample. This is sometimes called the absolute contribution approach. On the other hand, there is a level of release to the environment with a resultant corresponding level in food, air and water below which it is absurd to perform any specific isotopic analyses. Consequently, the sensitivities outlined in Table 4.11.2 are a compromise between the preceding two requirements. This schedule will insure that changes in the environmental radioactivity can be detected. The materials which first show changes in radioactivity are sampled most frequently. Those which are less affected by transient changes but show long-term accumulations are sampled less frequently. After a few years of operation, it will be desirable to review the established limits. Data on the actual concentrations in food or other organisms (if any are observed) will permit this reevaluation.

The lake bottom sediment samples taken twice during the summer months will be sufficient to detect any buildup in Cs-137.

References

FSAR, Section 2.5

Table 4.11.2

Sensitivity Requirements

Sample Class	Avg Total Sample Size	Sensitivity Requirements μCi/ml: Gross β	I^{131}	Fe^{59}, Co^{60}, Zn^{65}	Cs^{137}	Sr^{90}	$Ba\text{-}La^{140}$	Unidentified β Decision Limit
Air	2×10^{8} ml	1×10^{-14}	1×10^{-12}	-	1×10^{-12}	1×10^{-13}	1×10^{-12}	1×10^{-13}
*Water	1000 ml	5×10^{-9}	2×10^{-8}	1×10^{-7}	1×10^{-7}	1×10^{-9}	-	1×10^{-8}
Organic	1000 gm	1×10^{-8}	2×10^{-7}	-	2×10^{-7}	1×10^{-8}	-	5×10^{-8}
Milk	1000 ml	-	3×10^{-8}	-	3×10^{-8}	1×10^{-9}	3×10^{-8}	-
Film	-	-	-	10 mRem/Month Gross β, γ Above Background	-	-	-	-
or								
TLD	-	-	-	10 mRem/Month Gross γ Above Background	-	-	-	-

Preoperational Survey Only

*In addition to the water analyses, tritium analyses will be required at a sensitivity level of 1×10^{-7} μCi/ml on all Lake Michigan water samples.

5.0 DESIGN FEATURES

5.1 SITE

The Palisades reactor shall be located on 487 acres owned by Consumers Power Company on the eastern shore of Lake Michigan approximately four and one-half miles south of the southern city limits of South Haven, Michigan. Figure II-3 of the Palisades FSAR shows the plan of the site. The minimum distance to the boundary of the exclusion area as defined in 10 CFR 100.3 shall be 677 meters.

5.2 CONTAINMENT DESIGN FEATURES

5.2.1 Containment Structure

a. The containment structure completely encloses the primary coolant system to minimize release of radioactive material to the environment should a failure of the primary coolant system occur. The prestressed, post-tensioned concrete structure provides adequate biological shielding for both normal operation and accident situations and is designed for low leakage at a design pressure of 55 psig and 283°F.

The principal design basis for the structure is that it be capable of withstanding the internal pressure resulting from a design basis loss-of-coolant accident. In this event, the total energy contained in the water of the primary coolant system is assumed to be released into the containment through a double-ended break of the largest primary coolant pipe coincident with a loss of normal and standby electrical power. Subsequent pressure behavior is determined by the engineered safety features and the combined influence of energy sources and heat sinks.

b. The external design pressure of the containment shell is 3 psig. This value is approximately 0.5 psig greater than the maximum external pressure that could be developed if the containment were sealed during a period of low barometric pressure and high temperature and, subsequently, the containment atmosphere were cooled with a concurrent major rise in barometric pressure. Vacuum breakers are therefore not provided.

c. The containment is designed as a seismic Class I structure.

5.2 CONTAINMENT DESIGN FEATURES (Contd)

5.2.2 Penetrations

a. All penetrations through the steel-lined concrete structure for electrical conductors, pipe, ducts, air locks and doors are of the double-barrier design.

b. The automatically actuated containment isolation valves are designed to close upon high radiation or high pressure in the containment structure. No single component failure in the actuation system will prevent the isolation valves from functioning as designed.

5.2.3 Containment Structure Cooling Systems

a. The containment air cooling system includes four separate self-contained units which cool the containment air during normal operation and limit the pressure rise in the event of a design accident. Three units, each with a cooling water flow of 4875 gpm with an inlet temperature of $75^{o}F$, will remove 229×10^{6} Btu/hr of heat.

b. The containment spray system is capable of removing 233×10^{6} Btu/hr (two pumps) from the containment atmosphere at $283^{o}F$ by spraying the water from the 270,000-gallon SIRW tank. Recirculation of spray water from the containment sump through heat exchangers into the containment atmosphere is also provided.

Under this mode of operation, the heat removal capability is 167×10^{6} Btu/hr based upon 4000 gpm of component cooling water flow with $114^{o}F$ inlet temperature through the heat exchanger and 1420 gpm of spray water flow at $283^{o}F$ inlet temperature.

5.3 NUCLEAR STEAM SUPPLY SYSTEM (NSSS)

5.3.1 Primary Coolant System

a. The primary coolant system shall be designed and constructed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Vessels, including all addenda through the winter of 1965; and the ASA Code for Pressure Piping B31.1.

b. The primary coolant system shall be designed for a pressure of 2500 psia and a temperature of $650^{o}F$ except for the pressurizer which shall have a design temperature of $700^{o}F$.

c. The volume of the primary coolant system shall be approximately 10,900 cubic feet.

5.3 NUCLEAR STEAM SUPPLY SYSTEM (NSSS) (Contd)

5.3.2 Reactor Core and Control

a. The reactor core shall approximate a right circular cylinder with an equivalent diameter of 136.71 inches and an active height of 132 inches.

b. The reactor core shall consist of approximately 43,000 Zircaloy-4 clad fuel rods containing slightly enriched uranium in the form of sintered UO_2 pellets. The fuel rods shall be grouped into 204 bundles arranged for a three-batch refueling as shown in Figure 3-2 of the Palisades FSAR. A core plug or plugs may be used to replace one or more fuel bundles subject to the analysis of the resulting power distribution. The initial enrichments of the fuel bundles are:

Batch A - 1.65 W/O
Batch B - 2.08 and 2.54 W/O
Batch C - 2.54 and 3.20 W/O

c. The fully loaded core shall contain approximately 211,000 pounds UO_2 and approximately 56,000 pounds of Zircaloy-4. A small amount of alumina-boron carbide poison shall be placed in the fuel bundles for long-term reactivity control as shown in Figure 3-2 of the FSAR.

d. The core excess reactivity shall be controlled by a combination of boric acid chemical shim, cruciform control rods, and mechanically fixed boron rods where required. Forty-five control rods shall be distributed throughout the core as shown in Figure 3-5 of the FSAR. Four of these control rods may consist of part-length absorbers.

5.3.3 Emergency Core Cooling System

An emergency core cooling system shall be installed consisting of various subsystems each with internal redundancy. These subsystems shall include four safety injection tanks, three high-pressure and two low-pressure safety injection pumps, a safety injection and refueling water storage tank, and interconnecting piping as shown in Section 6 of the FSAR.

5.4 FUEL STORAGE

5.4.1 New Fuel Storage

a. Unirradiated fuel bundles will normally be stored in the dry new fuel storage rack with an effective multiplication factor of less

5.4 FUEL STORAGE (Contd)

than 0.7. The open grating floor below the rack and the covers above the racks, along with generous provision for drainage, precludes flooding of the new fuel storage rack.

b. New fuel may also be stored in shipping containers.

c. New fuel may also be stored in the spent fuel pool racks which have an effective multiplicator of 0.95 with new Palisades Batch C fuel and unborated water.

d. The new fuel storage racks are designed as a Class I structure.

5.4.2 Spent Fuel Storage

a. Irradiated fuel bundles will be stored, prior to off-site shipment in the stainless steel-lined spent fuel pool.

b. The spent fuel racks are designed to maintain fuel in a geometry which insures an effective multiplication factor of 0.95 or less with new fuel flooded with unborated water.

c. The spent fuel pool is normally filled with borated water with a concentration of approximately 1720 ppm.

d. The spent fuel racks are designed as a Class I structure.

References

FSAR, Appendix A.

FSAR, Appendix B.

6.0 ADMINISTRATIVE CONTROLS

6.1 ORGANIZATION, REVIEW AND AUDIT

6.1.1 The Plant Superintendent is directly responsible for the safe operation of the facility.

6.1.2 In all matters pertaining to operation of the plant and to these Technical Specifications, the Plant Superintendent shall report to and be directly responsible to the Electric Production Superintendent - Nuclear or equivalent. The organization is shown in Figure 6-1.

6.1.3 Organization for conduct of operations of the plant is shown in Figure 6-2.

6.1.4 Minimum qualifications for supervisory and professional personnel on the Plant Staff shall be in accordance with Section 4 of the "Proposed Standard for Selection and Training of Personnel for Nuclear Power Plants," prepared by the ANS-3 Committee, Draft No 9 dated 7-3-69.

6.1.5 The supervisory and professional personnel on the Plant Staff shall be retrained in accordance with Section 5.5 of the "Proposed Standard for Selection and Training of Personnel From Nuclear Power Plants," prepared by the ANS-3 Committee, Draft No 9 dated 7-3-69.

6.1.6 Until such time as the start-up tests and warranty run are completed, and sufficient "hot" operator license examinations have been successfully completed to provide the shift staffing as indicated in Figure 6-2, each operating shift shall consist of at least five persons, including at least one licensed senior reactor operator, one licensed reactor operator, and one or more senior staff members from either the Plant Staff, qualified members of the General Office Staff, or NSSS vendor's staff or consultants, who, by virtue of their training and experience, can provide competent technical support for the start-up and power ascension program.

6.1.7 Plant Review Committee (PRC)

A Plant Review Committee shall be constituted and have the responsibilities and authorities as outlined below:

a. Membership

(1) Chairman: Plant Superintendent, or designated alternate.

(2) Assistant Plant Superintendent.

(3) Technical Engineer.

(4) Reactor Engineer.

(5) Instrument and Control Engineer.

6.1 ORGANIZATION, REVIEW AND AUDIT (Contd)

(6) Senior Engineer or General Engineer.

(7) Chemical and Radiation Protection Engineer.

(8) Shift Supervisor (one).

(9) Plant Health Physicist.

b. The qualifications of the regular members of the Plant Review Committee (with regard to the combined experience and technical specialties of the individual members) shall be maintained at a level at least equal to or higher than those described in 6.1.4.

c. Meeting Frequency: Monthly, and as required on call of the Chairman.

d. Quorum: Chairman plus four members.

e. Responsibilities

(1) Review all proposed tests and experiments and the results thereof when applicable.

(2) Review changes to equipment or systems having safety significance, or which may constitute "an unreviewed safety question," pursuant to 10 CFR 50.59.

(3) Investigate reported or suspected violations of Technical Specifications or internal rules, procedures or regulations, to include reporting, evaluations and recommendations to prevent recurrence.

(4) Review proposed changes to Technical Specifications or licenses.

(5) Review abnormal performance of plant equipment and operating anomalies.

(6) Review unusual or abnormal occurrences.

(7) Review procedures as specified in Section 6.4 of this document.

(8) Perform special reviews and investigations and render reports thereon as requested by the Chairman of the Safety Audit and Review Board.

(9) Review plant operation to detect potential safety hazards.

f. Authority

(1) The Plant Review Committee shall be advisory.

6.1 ORGANIZATION, REVIEW AND AUDIT (Contd)

(2) The Plant Review Committee shall recommend to the Plant Superintendent approval or disapproval of proposals under Items e(1), (2), (4), and (7) above.

In the event of disagreement between the recommendations of the Plant Review Committee and the actions contemplated by the Plant Superintendent, the course determined by the Plant Superintendent to be the more conservative will be followed with immediate notification to the Chairman of the Safety Audit and Review Board.

(3) The Plant Review Committee shall make tentative determinations as to whether or not proposals considered by the committee involve unreviewed safety questions. This determination shall be subject to review and approval by the Safety Audit and Review Board.

g. Records

Minutes shall be kept of all meetings of the Plant Review Committee and copies shall be sent to the Chairman of the Safety Audit and Review Board.

6.1.8 Safety Audit and Review Board

a. Membership

(1) The Chairman, Vice-Chairman and board members are appointed by the Electric Production Superintendent - Nuclear, subject to the approval of the Senior Vice President - Electric Operations, and may include an outside member if deemed desirable. The minimum membership of the board shall be five (5) and the maximum ten (10).

(2) The minimum qualifications of the Safety Audit and Review Board with regard to the individual members shall be those specified in Section 4.6 of the "Proposed Standard for Selection and Training of Personnel for Nuclear Plants," prepared by the ANS-3 committee, Draft No 9 dated 7-3-69, and shall collectively provide expertise in:

(a) Reactor operations.

(b) Reactor engineering.

6.1 ORGANIZATION, REVIEW AND AUDIT (Contd)

(c) Chemistry and radiochemistry.

(d) Metallurgy and radiation damage.

(e) Instrumentation and control systems.

(f) Radiological safety.

(g) Mechanical and electrical systems.

(h) Any other appropriate field required by the unique characteristics of the facility.

b. Meeting Frequency

At least three times per year at intervals not to exceed five months and as required, on call of the Chairman.

c. Quorum: Chairman or Vice-Chairman plus four members.

d. Responsibilities

(1) Review proposed changes to the operating license including Technical Specifications.

(2) Review minutes of meetings of the Plant Review Committee to determine if matters considered by that committee involve unreviewed or unresolved safety questions.

(3) Review matters including proposed changes or modifications to plant systems or equipment having safety significance or referred to it by the Plant Review Committee or by the Plant Superintendent.

(4) Conduct periodic audits of plant operations.

(5) Investigate all reported instances of violations of Technical Specifications, reporting findings and recommendations to prevent recurrence to the Senior Vice President - Electric Operations.

(6) Perform special reviews and investigations and render reports thereon as requested by the Senior Vice President - Electric Operations.

(7) Review proposed tests and experiments having safety significance and results thereof when applicable.

(8) Review abnormal performance of plant equipment and anomalies.

(9) Review unusual occurrences which are reportable under provisions 10 CFR 20 and 10 CFR 50.

6.1 ORGANIZATION, REVIEW AND AUDIT (Contd)

(10) Review of occurrences if safety limits are exceeded.

(11) Evaluate actions taken under 6.1.7 f(2).

e. Authority

(1) Approve proposed changes to the operating license including Technical Specifications and revised bases for submission to the AEC.

(2) Approve proposed changes or modifications to plant systems or equipment, provided such changes or modifications do not involve unreviewed safety questions.

(3) Recommend to the Company Senior Vice President - Electric Operations appropriate action to prevent recurrence of any violations of Technical Specifications.

f. Records

Minutes shall be recorded of all meetings of this committee. Copies of the minutes shall be forwarded to the Senior Vice President - Electric Operations, Plant Superintendent and such others as the Chairman may designate.

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

6.2.1 Any abnormal occurrence shall be reported immediately to the Electric Production Superintendent - Nuclear and promptly reviewed by the Plant Review Committee.

6.2.2 The Plant Review Committee shall prepare a separate report for each abnormal occurrence. This report shall include an evaluation of the cause of the occurrence and also recommendations for appropriate action to prevent or reduce the probability of a recurrence.

6.2.3 Copies of all such reports shall be submitted to the Electric Production Superintendent - Nuclear and to the Chairman of the Safety Audit and Review Board for review and approval of any recommendation.

6.2.4 The Electric Production Superintendent - Nuclear shall promptly report the circumstances of any abnormal occurrence to the AEC when appropriate.

CONSUMERS POWER COMPANY
Palisades Plant
Management Organization Chart

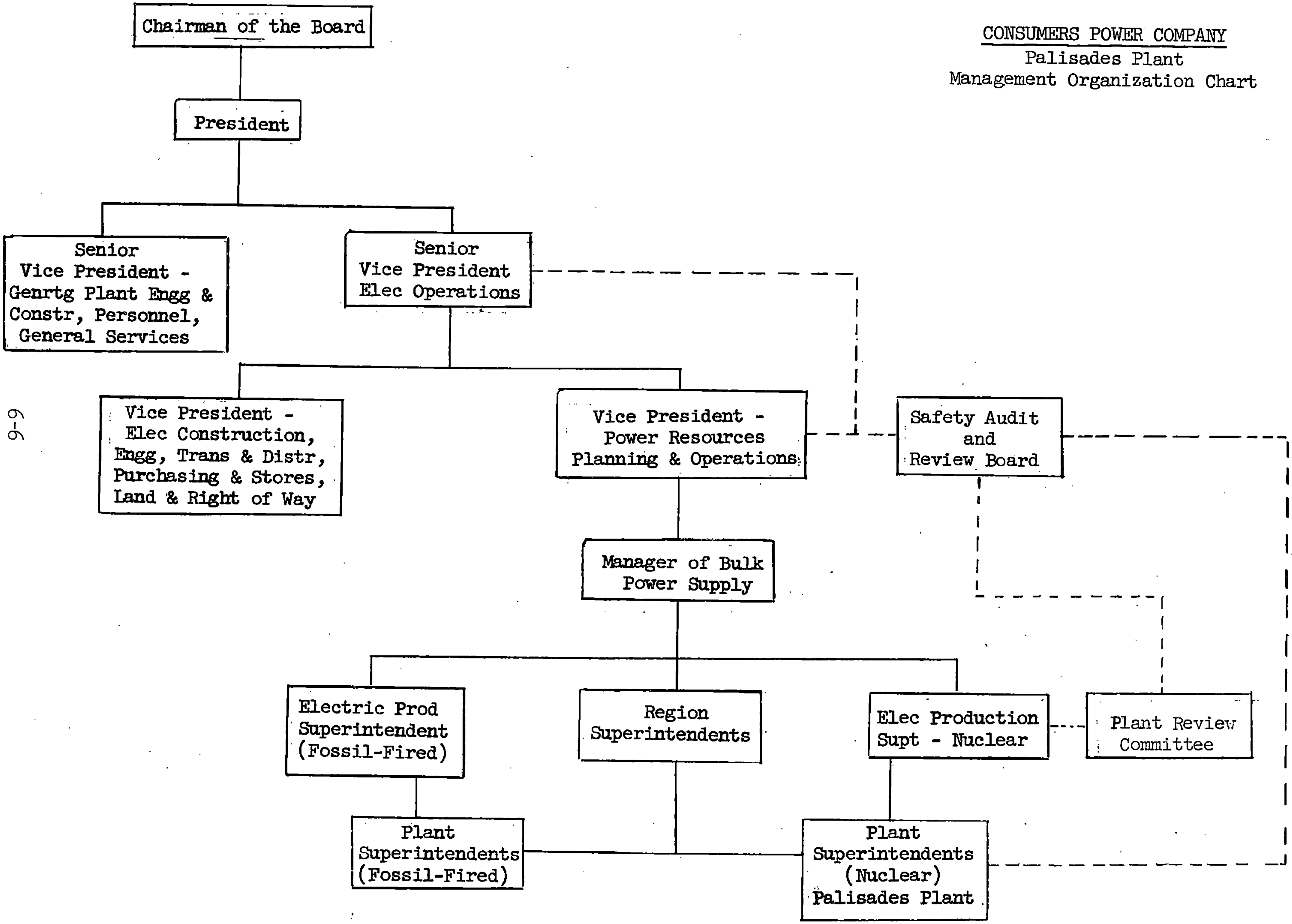


Figure 6-1

CONSUMERS POWER COMPANY
Palisades Plant
Conduct of Operations Chart

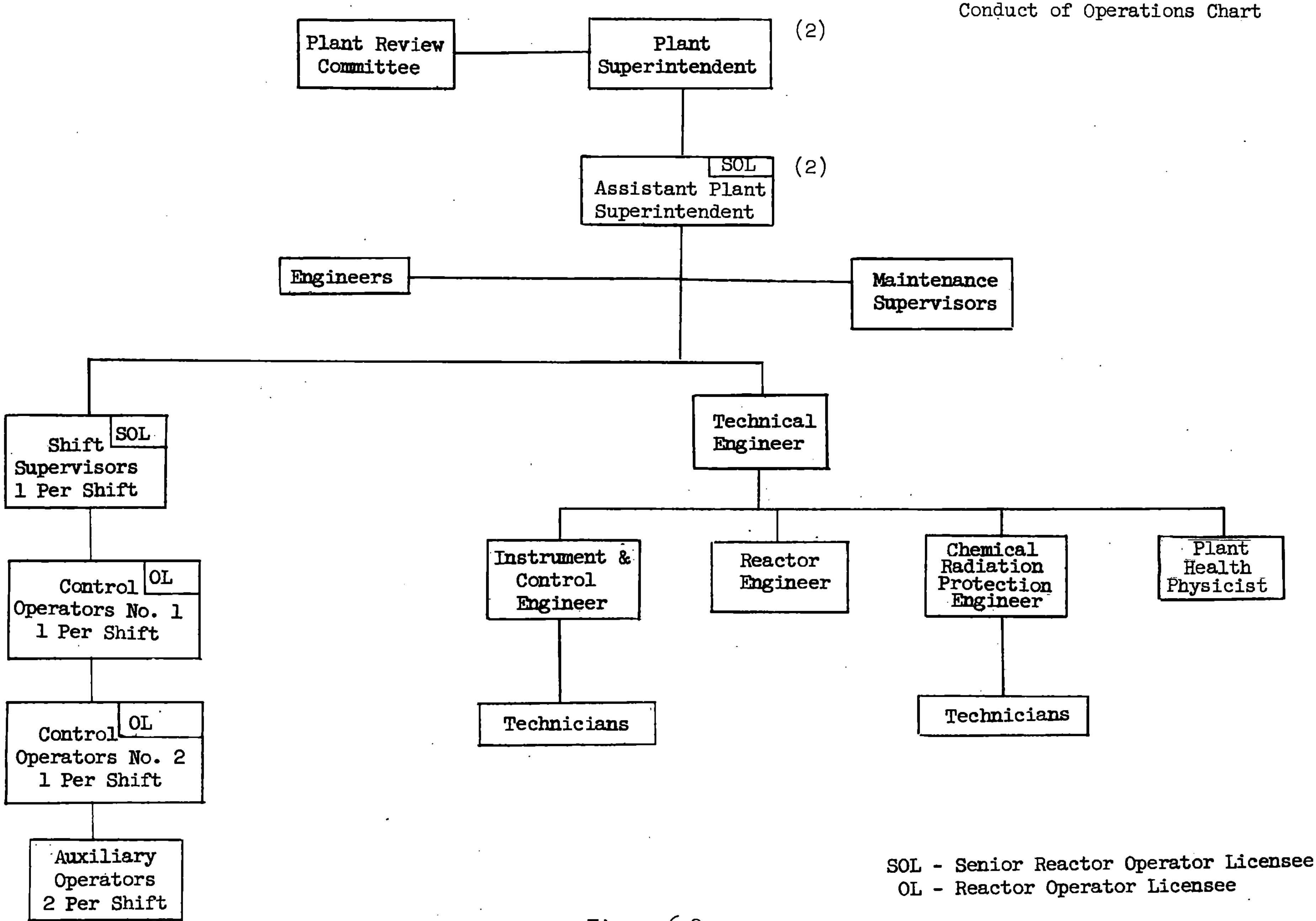


Figure 6-2

Figure 6-2 (Contd)

CONDUCT OF OPERATIONS CHART NOTES

1. The following requirements are in addition to the applicable regulations pursuant to Section 50.54(m) of 10 CFR 50.
   a. Two licensed reactor operators shall be in the control room during start-up and shutdown of the reactor and during recovery from reactor trips.
   b. A licensed senior reactor operator shall be on site at all times fuel is in the reactor vessel.
   c. The shift complement shall consist of at least five persons, including one licensed senior reactor operator and two licensed reactor operators except as specified in Technical Specification 6.1.6.
   d. A licensed operator shall be in the control room whenever fuel is in the reactor.
2. The Plant Shift Supervisor shall report to a supervisor having a Senior Reactor Operator's License. This supervisor shall be the Assistant Plant Superintendent or Plant Superintendent.

6.3 ACTION TO BE TAKEN IF A SAFETY LIMIT IS EXCEEDED

6.3.1 If a safety limit is exceeded, the reactor shall be shut down and reactor operation shall not be resumed until approval is received from the AEC.

6.3.2 The occurrence shall be reported immediately to the Electric Production Superintendent - Nuclear and to the Chairman of the Safety Audit and Review Board.

6.3.3 The Electric Production Superintendent - Nuclear shall promptly report the circumstances to the AEC, as specified in Section 6.6.1.

6.3.4 A complete analytical report of the circumstances leading up to and resulting from the occurrence, together with recommendations to prevent a recurrence, shall be prepared by the Plant Review Committee. This report shall be submitted to the Electric Production Superintendent - Nuclear and to the Chairman of the Safety Audit and Review Board. Appropriate analyses or reports will be submitted to the AEC.

6.4 UNIT OPERATING PROCEDURES

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

a. Normal start-up, operation, and shutdown of the complete plant and of all systems and components involving nuclear safety of the plant.
b. Specific malfunctions of systems or components, including alarms and abnormal reactivity changes.
c. Emergency conditions involving possible or actual release of radioactive materials.
d. Preventative or corrective maintenance operations which could have an effect on the safety of the reactor.
e. Refueling operations.
f. Radiation protection procedures.

6.4.2 All procedures described in 6.4.1, and changes thereto, shall be reviewed by the Plant Review Committee and approved by the Plant Superintendent prior to implementation, except as provided in 6.4.3.

6.4.3 Temporary changes to procedures which do not change the intent of the original procedures may be made, provided such changes are approved by two members of the Plant Review Committee, at least one of whom shall

6.4 UNIT OPERATING PROCEDURES (Contd)

be the Shift Supervisor. Such changes shall be documented and subsequently reviewed by the Plant Review Committee and approved by the Plant Superintendent.

6.4.4 Periodic drills shall be conducted on Site Emergency Procedures including assembly preparatory to evacuation of site and a check of adequacy of communications with off-site support groups.

6.4.5 Radiological Controls

a. Written information shall be prepared and maintained for all plant personnel regarding permissible radiation exposure limits, and procedures for control of radiation exposure.

b. The radiation protection controls are intended to meet the requirements of 10 CFR 20 with the following exceptions:

(1) Section 20.103 - "Exposure of individuals to concentrations of radioactive materials in restricted area...."

Allowance can be made for the use of respiratory protective equipment in conjunction with activities authorized by the operating license for this plant in determining whether individuals in restricted areas are exposed to concentrations in excess of the limits specified in Appendix B, Table 1, Column 1, of 10 CFR 20, subject to the following conditions and limitations:

(a) The individual uses respiratory or other appropriate protective equipment such that the total intake, in any period of seven consecutive days by inhalation, ingestion or absorption, would not exceed that intake specified by the continuous concentration limits listed in Appendix B, Table 1, Column 1, of 10 CFR 20 for a period of 40 hours.

(b) Each respirator user shall be advised that he may leave the area 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.

(c) A respiratory protection program, described in the FSAR, Section 11.2.6.9, adequate to insure that the objectives of Item (a) above are met, shall be maintained.

6.4 UNIT OPERATING PROCEDURES (Contd)

6.4.6 Average Concentrations

The average concentrations of airborne radioactivity of plant origin in areas normally occupied by employees, are expected to be considerably less than the applicable MPC, established in Appendix B, Table 1, of 10 CFR, Part 20. Some areas within the plant not normally occupied could be expected to have airborne concentrations in excess of the applicable limit. These areas have to be entered by plant personnel at certain times in order to insure safe operation of the plant. Therefore, when personnel enter these areas, or other areas of high airborne radioactivity in excess of twice the applicable MPC, they will be required to wear appropriate respiratory protective equipment. Protection factors shall not be assigned in excess of those given in Table 6.4.1.

Table 6.4.1

Protection Factors for Respirators

Description	Modes[1]	Protection Factors[2]: Particulates, Vapors and Gases Except Tritium Oxide[3]	Protection Factors[2]: Tritium Oxide
I. Air-Purifying Respirators			
Facepiece, Half Mask		10	1
Facepiece, Full		100	1
II. Atmosphere-Supplying Respirators			
A. Air Line Respirator			
Facepiece, Full	CF	1000	2
Facepiece, Full	PD	1000	2
Hood	CF	1000	2
Suit	CF	(4)	(4)
B. Self-Contained Breathing Apparatus (SCBA)			
Facepiece, Full	R	1000	2
C. Combination Respirator			
Any combination of air-purifying and atmosphere-supplying respirator.		Protection factor for type and mode of operation as listed above.	

[1] CF: Continuous Flow
PD: Pressure Demand (ie, Always Positive Pressure)
R: Recirculating

[2] (a) For purposes of this authorization, 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 airborne concentration to determine the concentration inhaled by the wearer, according to the following formula:

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

Table 6.4.1

Protection Factors for Respirators (Contd)

(b) The protection factors apply:

(i) Only for individually fitted respirators worn by trained individuals and used and maintained under supervision in a well-planned respiratory protection program.

(ii) For air-purifying respirators only when high efficiency particulate filters and/or sorbents appropriate to the hazard are used.

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

(3) Excluding radioactive contaminants that present an absorption or submersion hazard.

(4) Appropriate protection factors must be determined taking account of the permeability of the suit to the contaminant under conditions of use. No protection factor greater than 1000 shall be used except as authorized by the Commission.

Note 1: Protection factors for respirators as may be approved in the future by the US Bureau of Mines according to approval schedules for respirators to protect against airborne radionuclides may be used in lieu of the protection factors listed in this table. Where additional respiratory hazards other than radioactive ones are present, especially those immediately dangerous to life, the selection and use of respirators shall also be governed by the approvals of the US Bureau of Mines in accordance with their applicable schedules.

Note 2: Radioactive contaminants for which the concentration values in Appendix B, Table 1, of 10 CFR, Part 20, are based on internal dose due to inhalation may, in addition, present external exposure hazards at higher concentrations.

6.5 PLANT OPERATING RECORDS

Records and logs relative to the following items shall be retained for six (6) years unless a longer period is required by applicable regulations. Operating charts for the first year's operation will be permanently stored.

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

b. Records of principal maintenance activities, including inspection, repair, substitution or replacement of principal items of equipment pertaining to nuclear safety.

c. Records of abnormal occurrences.

d. Records of periodic checks, inspections and calibrations performed to verify that surveillance requirements are being met.

*e. Records and prints of changes made to the plant as described in the Final Safety Analysis Report.

*f. Records of new and spent fuel inventory and assembly histories.

*g. Records of monthly plant radiation and contamination surveys.

*h. Records of off-site environmental monitoring surveys.

*i. Records of radiation exposure of all plant personnel, including all contractors and visitors to the plant who enter radiation control areas.

*j. Records of radioactivity in liquid and gaseous wastes released to the environment.

k. Records of any special reactor tests or experiments.

l. Records of changes made in the Operating Procedures.

*These items will be permanently retained.

6.6 PLANT REPORTING REQUIREMENTS

In addition to the reports required by AEC regulations, other reports to the AEC will be made as follows:

6.6.1 A notification shall be made to the Atomic Energy Commission within 24 hours (by telephone or telegraph to the Director of the appropriate AEC Regional Compliance office) of any of the following events:

a. Any significant variation of measured values in a nonconservative direction from corresponding predicted values of safety-connected operating parameters.

b. Incidents or conditions, excluding planned maintenance, relating to operation of a facility which prevented or could have prevented the performance, as described in the Technical Specifications, of those systems designed to prevent or mitigate the consequences of accidents.

c. Incidents or conditions which result in exceeding a safety limit established in the Technical Specifications.

d. Abnormal occurrences, as defined in these Technical Specifications.

6.6.2 A written report shall be submitted within 10 days to the Director, Division of Reactor Licensing, USAEC, with a copy to the Director of the appropriate AEC Regional Compliance office of any event previously reported under the provisions of 6.6.1 above. The report shall describe, analyze, evaluate the safety implications of the event, and outline the corrective actions and measures taken or planned to prevent recurrence.

6.6.3 A written report shall be submitted within 30 days to the Director, Division of Reactor Licensing, USAEC, with a copy to the Director of the appropriate AEC Regional Compliance office of the following events:

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

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

6.6.4 A written report shall be submitted to the Director, Division of Reactor Licensing, USAEC, (subsequent to receipt of the initial operating license or any amendment to the facility license involving the planned increase

in reactor power level or the installation of a new core) within 60 days after initial criticality under the new license conditions. This report shall describe the measured values of the new operating conditions or characteristics.

6.6.5 A written report shall be submitted at the end of each six months' period (Semiannual Report) to the Director, Division of Reactor Licensing, USAEC. Such reports are due within 60 days after the end of each reporting period.

The following information shall be provided:

a. Operations Summary

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

b. Power Generation

Tabulation of the thermal and electrical output of the plant during the reporting period, and the cumulative total outputs since initial criticality, including:

(1) Gross thermal power generated (in MWh).
(2) Gross electrical power generated (in MWh).
(3) Net electrical power generated (in MWh).
(4) Number of days during which the reactor was operated.
(5) Number of hours the reactor was critical.
(6) Number of hours the generator was on line.
(7) Equivalent full-power hours.
(8) Histogram of thermal power vs time.

c. Shutdowns

Descriptive material covering all outages occurring during the reporting period, including a description of the corrective action taken for each unplanned outage. For each outage, provide information on:

(1) The cause of the outage.
(2) The method of shutting down the reactor (eg, trip or manually controlled deliberate shutdown).
(3) Duration of the outage in hours.
(4) Plant status (eg, cold shutdown or hot standby) during the outage.

6.6 PLANT REPORTING REQUIREMENTS (Contd)

d. Maintenance

Discussion of electrical, mechanical and general maintenance performed during the report period and having potential effects on the safety of the facility. The specific systems involved shall be identified and information shall be provided on:

(1) The nature of the maintenance (eg, routine, emergency, preventive or corrective).

(2) The effect, if any, on the safe operation of the reactor.

(3) The cause of any malfunction for which emergency or corrective maintenance was required.

(4) The results of any such malfunction.

(5) The corrective and preventive action taken to preclude recurrence.

(6) The time required for completion.

e. In-Service Surveillance Program

Description of the in-service inspections performed during the reporting period.

f. Facility Changes, Tests and Experiments

A summary description of changes in the facility or in procedures and of tests and experiments carried out under the conditions of Section 50.59(a) of 10 CFR 50. (Records shall be kept and a written safety evaluation shall be made of all changes, tests and experiments performed that do not require prior Commission approval. Under the conditions of Section 50.59(b), 10 CFR 50, a brief description of the change and a summary of the safety evaluation will be submitted to the Commission as a part of the Semiannual Report (Section 6.6.5).)

g. Radioactive Effluent Releases

The quantities of liquid, gaseous and solid radioactive effluents released from the facility and the environmental monitoring shall be provided as specified below and reported annually by March 31 for the preceding calendar year:

(1) Liquid Wastes (Summarized on a Monthly Basis)

(a) Total number of gross curies other than tritium released during the reporting period.

(b) Total number of curies of tritium released during the reporting period.

6.6 PLANT REPORTING REQUIREMENTS (Contd)

(c) The MPC used and the isotopic composition, if greater than:

1. 1×10^{-7} μCi/cc for fission and activation products (except tritium).
2. 3×10^{-3} μCi/cc for tritium (only if the average concentration for monthly releases exceeds this value).

(d) Total estimated annual radioactivity (in curies) released by nuclide (other than tritium) based on representative isotopic analysis.

(e) Annual average concentration at point of release (in μCi/cc).

(f) Total volume (in gallons) of effluent water (including diluent) during periods of release.

(2) Gaseous Wastes (Summarized on a Monthly Basis)

(a) Noble gases radioactivity discharged during the year (in curies).

(b) Halogens radioactivity with half-lives greater than eight days discharged during the year (in curies).

(c) Particulates radioactivity with half-lives greater than eight days discharged during the year (in curies).

(d) The MPC used if greater than:

1. 3×10^{-8} μCi/cc for noble gases.
2. 1×10^{-10} μCi/cc for (b) above.
3. 3×10^{-11} μCi/cc for (c) above.

(e) Estimated annual radioactivity discharged during the year (in curies) released by each halogen or particulate nuclide, based on representative isotopic analysis.

(3) Solid Wastes

(a) The estimated total amount of solid waste stored on site (in cubic feet).

(b) The total activity involved (in curies).

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

6.6 PLANT REPORTING REQUIREMENTS (Contd)

(4) Environmental Monitoring

(a) A narrative summary of the results of off-site airborne environmental surveys performed during the report period as described in Specification 4.10.

(b) For each medium sampled during the year, a list of the sampling locations, the total number of samples, and the highest, lowest, and the average concentrations for the highest location.

(c) If levels of radioactive materials in environmental media due to plant origin indicate the likelihood of public exposure in excess of 5% of those that would result from continuous exposure to the concentration values listed in Appendix B, Table II, estimates of the likely resultant exposure to individuals and to population groups and assumptions on which estimates are made.

6.6.6 Special Reports

The following written reports shall be submitted to the Director, DRL, USAEC:

1. In the event a redundant component (or system) covered by these Technical Specifications is determined to be out of service for a period longer than those specified in other sections, it shall be the subject of a special maintenance report. This report shall be submitted within ten days of the above determination and shall describe:

   a. The nature of the problem and the specific steps to be taken to remedy the situation.

   b. An estimate of the time required to return the component (or system) to an operable condition.

   c. The amount of component (or system) redundancy remaining or the availability of other system(s) to perform the same function as the inoperable component (or system).

   d. Surveillance requirements on the operable component (or system).

6.6 PLANT REPORTING REQUIREMENTS (Contd)

2. Any significant changes of the information supplied in 6.6.6, 1.a, b, c, or d shall be reported within ten days.

3. Reports on the following areas shall be submitted as noted:

Area	Specification Reference	Submittal Date
Primary Containment Leak Rate Tests*	4.5	Upon Completion of Each Test
Prestressing, Anchorage, Liner and Penetration Tests	4.5.4 4.5.5	Upon Completion of Each Test
Primary System Surveillance Evaluation and Review	4.3	Five Years
In-Core Instrumentation Evaluation	3.1.1	Annually
Evaluation of Moderator Temperature Coefficient	3.1.2	Prior to Operation in Excess of 70% Rated Power
Initial Containment Structural Tests	4.5	Prior to Operation in Excess of 70% Rated Power
Tendon Surveillance Program Study	4.5.4	Within Six Months After Receipt of Initial Operating License

4. A comprehensive report presenting the results of the initial preoperational, start-up, power ascension and full-power test programs (including operating record of in-core and instrumentation) shall be submitted within one year of the commercial service date.

*Each integrated leak rate test of the primary containment shall be the subject of a summary technical report including results of the local leak rate tests since the last report. The report as described in the AEC Guide on Containment testing dated December 15, 1966, shall include data, analysis, and interpretations of the results which demonstrate compliance in meeting the specification leak rate limits.

6.6 PLANT REPORTING REQUIREMENTS (Contd)

5. An analysis and report shall be submitted to the Atomic Energy Commission on all surveillance specimens removed from the reactor vessel as described in Tables 4-15 and 4-17 of the FSAR. These reports shall include: (1) The information specified in ASTM E-185-66, "Recommended Practices for Surveillance Tests on Structural Materials in Nuclear Reactor"; (2) information obtained on the level of integrated fast neutron irradiation received by the specimens and the actual vessel material; and (3) the revised limitations on start-up, cooldown and operating conditions as designated by Technical Specifications Figures 3-1 and 3-2 and how these revised limitations were evaluated.