RANCHO SECO
nuclear generating station

TECHNICAL SPECIFICATIONS

UNIT ONE

SMUD

. TECHNICAL SPECIFICATIONS

These Technical Specifications apply to the Rancho Seco Nuclear Generating Station, Unit 1, and are in accordance with the requirements of 10 CFR 50, Section 50.36. The bases, which provide technical support or reference the pertinent FSAR section for technical support of the individual specifications, are included for informational purposes and to clarify the intent of the specification. These bases are not part of the Technical Specifications, and they do not constitute limitations or requirements for the licensee.

APPENDIX A

TO

TECHNICAL SPECIFICATIONS

FOR THE

RANCHO SECO UNIT 1

SACRAMENTO MUNICIPAL UTILITY DISTRICT

DOCKET NO. 50-312

Revised to incorporate NUREG-0472

May 1979

RANCHO SECO UNIT 1
TECHNICAL SPECIFICATIONS

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1. DEFINITIONS

The following terms are defined for uniform interpretation of these specifications.

1.1 RATED POWER

Rated power is a steady state reactor core output of 2772 MWt.

1.2 REACTOR OPERATING CONDITIONS

1.2.1 Cold Shutdown

The reactor is in the cold shutdown condition when it is subcritical by at least 1 percent $\Delta k/k$ and T_{avg} is no more than 200 F. Pressure is defined by Specification 3.1.2.

1.2.2 Hot Shutdown

The reactor is in hot shutdown condition when it is subcritical by at least 1 percent Δ k/k and T_{avg} is at or greater than 525 F.

1.2.3 Reactor Critical

The reactor is critical when the neutron chain reaction is self-sustaining and $K_{eff} = 1.0$.

1.2.4 Hot Standby

The reactor is in the hot standby condition when all of the following conditions exist:

A. T_{avg} is greater than 525 F.

B. The reactor is critical.

C. Indicated neutron power on the power range is less than 2% of the rated power.

1.2.5 Power Operation

The reactor is in a power operating condition when the indicated neutron power is above 2 percent of rated power as indicated on the power range channels.

1.2.6 Refueling Shutdown

The reactor is in the refueling shutdown condition when, even with all rods removed, the reactor would be subcritical by at least 1 percent $\Delta k/k$

1.2.6 (Continued)

and the coolant temperature at the decay heat removal pump suction is no more than 140 F. Pressure is defined by Specification 3.1.2. A refueling shutdown refers to a shutdown to replace or rearrange all or a portion of the fuel assemblies and/or control rods.

1.2.7 Refueling Operation

An operation involving a change in core geometry by manipulation of fuel or control rods when the reactor vessel head is removed.

1.2.8 Refueling Interval

Time between normal refuelings of the reactor, not to exceed 24 months for the first refueling and 18 months thereafter without prior approval of the NRC.

1.2.9 Startup

The reactor shall be considered on the startup mode when the shutdown margin is reduced with the intent of going critical.

1.2.10 Remain Critical

A technical specification that requires that the reactor shall not remain critical shall mean that an uninterrupted normal hot shutdown procedure will be completed with 12 hours.

1.2.11 T_{AVG}

At operating conditions T_{AVG} is defined as the arithmetic average of the coolant temperatures in the hot and cold legs of the loop with the greater number of reactor coolant pumps operating, if such a distinction of loops can be made.

1.2.12 Heatup - Cooldown Mode

The heatup-cooldown mode is the range of reactor coolant temperature greater than 200 F and less than 525 F.

1.3 OPERABLE

A component or system is operable when it is capable of performing its intended function within the required range. The component or system shall be considered to have this capability when: (1) it satisfies the limiting conditions for operation defined in Specification 3, and (2) it has been tested periodically in accordance with Specification 4, and has met its performance requirements.

1.4 PROTECTION INSTRUMENTATION LOGIC

1.4.1 Instrument Channel

An instrument channel is the combination of sensor, wires, amplifiers and output devices which are connected for the purpose of measuring the value of a process variable for the purpose of observation, control and/or protection. An instrument channel may be either analog or digital.

1.4.2 Reactor Protection System

The reactor protection system is shown in figures 7.1-1 and 7.2-2 of the FSAR. It is that combination of protective channels and associated circuitry which forms the automatic system that protects the reactor by control rod trip. It includes the four protection channels, their associated instrument channel inputs, manual trip switch, all rod drive control protective trip breakers and activating relays or coils.

1.4.3 Protection Channel

A protection channel, as shown in figure 7.1-1 of the FSAR (one of three or one of four independent channels, complete with sensors, sensor power supply units, amplifiers and bistable modules provided for every reactor protection safety parameter), is a combination of instrument channels forming a single digital output to the protection system's coincidence logic. Each protection channel includes two-key-operated bypass switches, a protection channel bypass switch and a shutdown bypass switch.

1.4.4 Reactor Protection System Logic

This system utilizes reactor trip module relays (coils and contacts) in all four of the protection channels as shown in figure 7.1-1 of the FSAR, to provide reactor trip signals for de-energizing the six control rod drive trip breakers. The control rod drive trip breakers are arranged to provide a one-out-of-two-times-two logic. Each element of the one-out-of-two-times-two logic is controlled by a separate two-out-of-four logic from the four reactor protection channels. With one channel bypassed and untripped, the two-out-of-four logic functions as a two-out-of-three logic for the three active channels.

1.4.5 Safety Features System Logic

This system utilizes relay contact output from individual channels arranged in three analog sub-systems and two two-out-of-three logic sub-systems as shown in figure 7.1-5 of the FSAR. The logic sub-system is wired to provide appropriate signals for the actuation of redundant safety features equipment on a two-of-three basis for any given parameter.

1.4.6 Degree of Redundancy

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

1.5 INSTRUMENTATION SURVEILLANCE

1.5.1 Trip Test

A trip test is a test of logic elements in a protection channel to verify their associated trip action.

1.5.2 Channel Test

A channel test is the injection of an internal or external test signal into the channel to verify its proper response, including alarm and/or trip initiating action, where applicable.

1.5.3 Instrument Channel Check

An instrument channel check is a verification of acceptable instrument performance by observation of its behavior and/or state; this verification includes comparison of output and/or state of independent channels measuring the same variable.

1.5.4 Instrument Channel Calibration

An instrument channel calibration is a test, and adjustment (if necessary), to establish that the channel output responds with acceptable range and accuracy to known values of the parameter which the channel measures or an accurate simulation of these values. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip and shall be deemed to include the channel test.

1.5.5 Heat Balance Check

A heat balance check is a comparison of the indicated neutron power and core thermal power.

1.5.6 Heat Balance Calibration

An adjustment of the power range channel amplifiers output to agree with the core thermal power as determined by a heat balance considering all heat losses and additions.

1.5.7 Source Check

A source check is the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

1.6 QUADRANT POWER TILT

Quadrant to average power tilt is expressed in percent as defined by the following equation:

$$100 \left(\frac{\text{Power in any core quadrant}}{\text{Average for all quadrants}} - 1 \right)$$

1.6.1 Reactor Power Imbalance

Reactor power imbalance is the power in the top half of the core minus the power in the bottom half of the core.

1.7 CONTAINMENT INTEGRITY

Containment integrity exists when the following conditions are satisfied:

A. The equipment hatch is closed and sealed and both doors of the personnel hatch and emergency hatch are closed and sealed except in B below.

B. At least one door on each of the personnel hatch and emergency hatch is closed and sealed during refueling operations or personnel passage through these hatches.

C. All non-automatic containment isolation valves and blind flanges are closed as required.

D. All automatic containment isolation valves are operable or closed in the safety features position.

1.7 (Continued)

E. The containment leakage satisfies Specification 4.4.1 and no known changes have occurred.

1.8 REPORTABLE OCCURRENCE

Defined under Administrative Controls Section 6.9.5.

1.9 TIME PERIODS

May be extended to a maximum of +25% to accomodate operations scheduling. The total maximum combined interval time for any three consecutive tests shall not exceed 3.25 times a single specified surveillance interval.

1.9.1 Shifts (S)

A time period covering at least once per twelve (12) hours.

1.9.2 Daily (D)

A time period spaced to occur at least once per twenty-four (24) hours.

1.9.3 Weekly (W)

A time period spaced to occur at least once per seven (7) days.

1.9.4 Fortnightly (F)

A time period covering two consecutive weeks spaced to occur 26 times a year.

1.9.5 Monthly (M)

A time period spaced to occur at least once per thirty-one (31) days.

1.9.6 Quarterly (Q)

A time period spaced to occur at least once per ninety-two (92) days.

1.9.7 Semi-Annually (S.Y.)

A time period spaced to occur at least once per six (6) months.

1.9.8 Annually (A)

A time period spaced to occur at least once per twelve (12) months.

1.9.9 Biennially (B.Y.)

A time period spaced to occur at least once in two (2) years.

1.9.10 Refueling Interval (R)

A time period spaced to occur at least once per eighteen (18) months.

1.9.11 Startup (S/U)

Prior to each reactor startup

1.9.12 Prior to Release (P)

Completed prior to each release

1.9.13 Not Applicable (NA)

Does not apply to stated condition.

1.10 SAFETY

Safety as used in these Technical Specifications shall mean nuclear safety and shall encompass all systems and components that have or may have an effect on the health and safety of the general public.

1.11 FIRE SUPPRESSION SYSTEMS

1.11.1 Water Fire Suppression System

The system shall consist of water sources, pumps and distribution piping with associated sectionalizing control of isolation valves. Such valves include yard hydrant valves and the first valve ahead of the water flow alarm device on each sprinkler header, hose standpipe or spray system riser which protect nuclear safety components.

1.11.2 Carbon Dioxide Fire Suppression System

The system shall consist of a CO_2 source and distribution piping with sectionalizing control valves which protect nuclear safety components.

1.12 STAGGERED TEST BASES

A STAGGERED TEST BASES shall consist of:

a. A test schedule for n systems, subsystems, trains or designated components obtained by dividing the specified test interval into n equal subintervals.

b. The testing of one system, subsystem, train or designated components during each subinterval.

1.13 PROCESS CONTROL PROGRAM

A PROCESS CONTROL PROGRAM (PCP) shall be the manual detailing the program of sampling, analysis, and evaluation within which SOLIDIFICATION of radioactive wastes from liquid systems is assured.

1.14 SOLIDIFICATION

Solidification shall be the conversion of liquid radioactive wastes to an immobilized free-standing solid.

1.15 OFFSITE DOSE CALCULATION MANUAL (ODCM)

An OFFSITE DOSE CALCULATION MANUAL (ODCM) shall be a manual containing the methodology and parameters to be used in the calculation of offsite dose due to radioactive gaseous and liquid effluents and in the calculation of gaseous and liquid effluent monitoring instrumentation alarm/trip setpoints and specific details of the environmental radiological monitoring program.

1.16 RESTRICTED AREA

That portion of the site property, the access to which is controlled by security fencing, equipment and personnel.

1.17 SITE BOUNDARY

The boundary of the SMUD owned property.

1.18 DOSE EQUIVALENT I-131

1.5 The DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites".

1.19 MEMBER(S) OF THE PUBLIC

MEMBER(S) OF THE PUBLIC shall include all individuals who by virtue of their occupational status have no formal association with the plant. This category shall include non-employees of the licensee who are permitted to use portions of the site for recreational, occupational, or other purposes not associated with plant functions. This category shall not include non-employees such as vending machine servicemen or postmen who, as part of their formal job function, occasionally enter an area that is controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials.

2. SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS, REACTOR CORE

Applicability

Applies to reactor thermal power, reactor power imbalance, reactor coolant system pressure, coolant temperature, and coolant flow during power operation of the plant.

Objective

To maintain the integrity of the fuel cladding.

Specification

2.1.1 The combination of the reactor system pressure and coolant temperature shall not exceed the safety limit as defined by the locus of points established in Figure 2.1-1. If the actual pressure/temperature point is within the restricted region the safety limit is exceeded.

2.1.2 The combination of reactor thermal power and reactor power imbalance (power in the top half of the core minus the power in the bottom half of the core expressed as a percentage of the rated power) shall not exceed the safety limit as defined by the locus of points (solid line) for the specified flow set forth in Figure 2.1-2. If the actual-reactor-thermal-power/reactor-power-imbalance point is above the line for the specified flow, the safety limit is exceeded.

Bases

The safety limits presented have been generated using BAW-2 critical heat flux (CHF) correlation (1) and the actual measured flow rate (2). This development is discussed in the Rancho Seco Unit 1, Cycle 2 Reload Report, reference (2). The flow rate utilized is 104.9 percent of the design flow (369,600 gpm) based on four-pump operation (2, 3).

To maintain the integrity of the fuel cladding and to 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

2.1 (Continued)

Bases (Continued)

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 neutron power, reactor coolant flow, temperature, and pressure can be related to DNB through the use of the BAW-2 correlation (1).

The BAW-2 correlation has been developed to predict DNB and the location of DNB for axially uniform and non-uniform 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 1.30. A DNBR of 1.30 corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur; this is considered a conservative margin to DNB for all operating condition. The difference between the actual core outlet pressure and the indicated reactor coolant system pressure has been considered in determining the core protection safety limits. The difference in these two pressures in nominally 45 psi; however, only 30 psi drop was assumed in reducing the pressure trip setpoints to correspond to the elevated location where the pressure is actually measured.

The curve presented in Figure 2.1-1 represents the conditions at which a minimum DNBR of 1.30 is predicted for the maximum possible thermal power (112 percent) when four reactor coolant pumps are operating (minimum reactor coolant flow is 104.9 percent of 369,600 gpm). This curve is based on the combination of nuclear power peaking factors, with potential effects of fuel densification and rod bowing, which result in a more conservative DNBR than any other shape that exists during normal operation.

The curves of Figure 2.1-2 are based on the more restrictive of two thermal limits and include the effects of potential fuel densification and rod bowing:

1. The 1.30 DNBR limit produced by the combination of the radial peak, axial peak and position of the axial peak that yields no less than a 1.30 DNBR.

2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limit is 20.4 KW/ft.

Power peaking is not a directly observable quantity and therefore limits have been established on the bases of the reactor power imbalance produced by the power peaking.

2.1 (Continued)

Bases (Continued)

The specified flow rates for Curves 1, 2, and 3 of Figure 2.1-2 correspond to the expected minimum flow rates with four pumps, three pumps, and one pump in each loop respectively.

The curve of figure 2.1-1 is the most restrictive of all possible reactor coolant pump-maximum thermal power combinations shown in figure 2.1-3. The curves of figure 2.1-3 represent the conditions at which a minimum DNBR of 1.3 is predicted at the maximum possible thermal power for the assumed design flow, or the local quality at the point of minimum DNBR is equal to 15 percent (3), whichever condition is more restrictive.

For Figure 2.1-3, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.30. The 1.30 DNBR curve for four-pump operation is more restrictive than any other reactor coolant pump situation because any pressure/temperature point above and to the left of the four-pump curve will be above and to the left of the other curves.

The maximum thermal power for three-pump operation depicted in Figure 2.1-2 is 88.65 percent due to a power level trip produced by the flux-flow ratio 1.06 times 74.4 percent design flow = 78.86 percent power plus the maximum calibration and instrumentation error. The maximum thermal power for other coolant pump conditions is produced in a similar manner. The actual maximum power levels are calculated by the RPS and will be directly proportional to the actual flow during partial pump operation. A thermal margin credit equivalent to 1% DNBR to offset the rod bowing penalty has been used as a result of the flow area (ditch) reduction factor included in the hot-channel thermal-hydraulic analysis. The 1% DNBR credit was approved (Reference 4) and is the only credit applied to offset the rod bow penalty.

References

(1) Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water, BAW-10000, March, 1970.

(2) Rancho Seco Unit 1, Cycle 2 Reload Report BAW.

(3) Rancho Seco Unit 1, Cycle 3 Reload Report BAW-1499, September, 1978.

(4) D. F. Ross and D. G. Eisenhut memorandum for D. B. Vassals and K. R. Goller, "Interim Safety Evaluation Report on the Effects of Fuel Rod Bowing on Thermal Margin Calculations for Light Water Reactors", dated December 9, 1976.

Figure 2.1-1. Core Protection Safety Limit, Pressure Vs Temperature

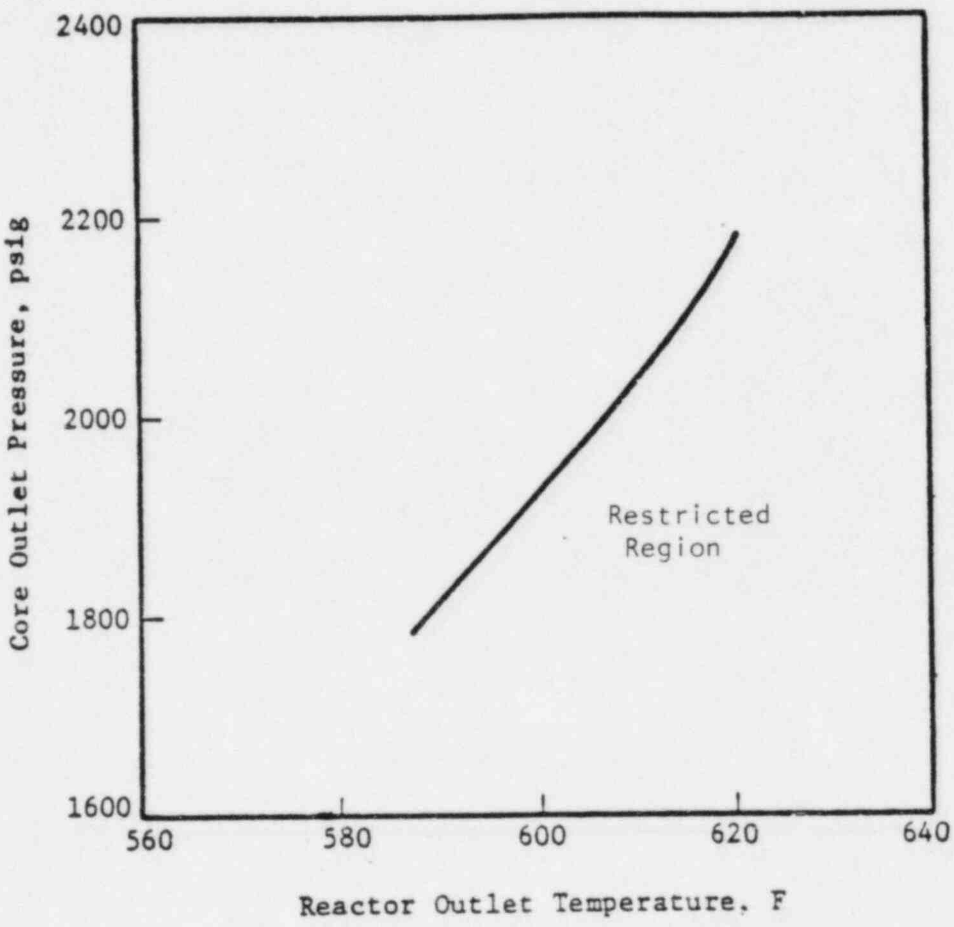
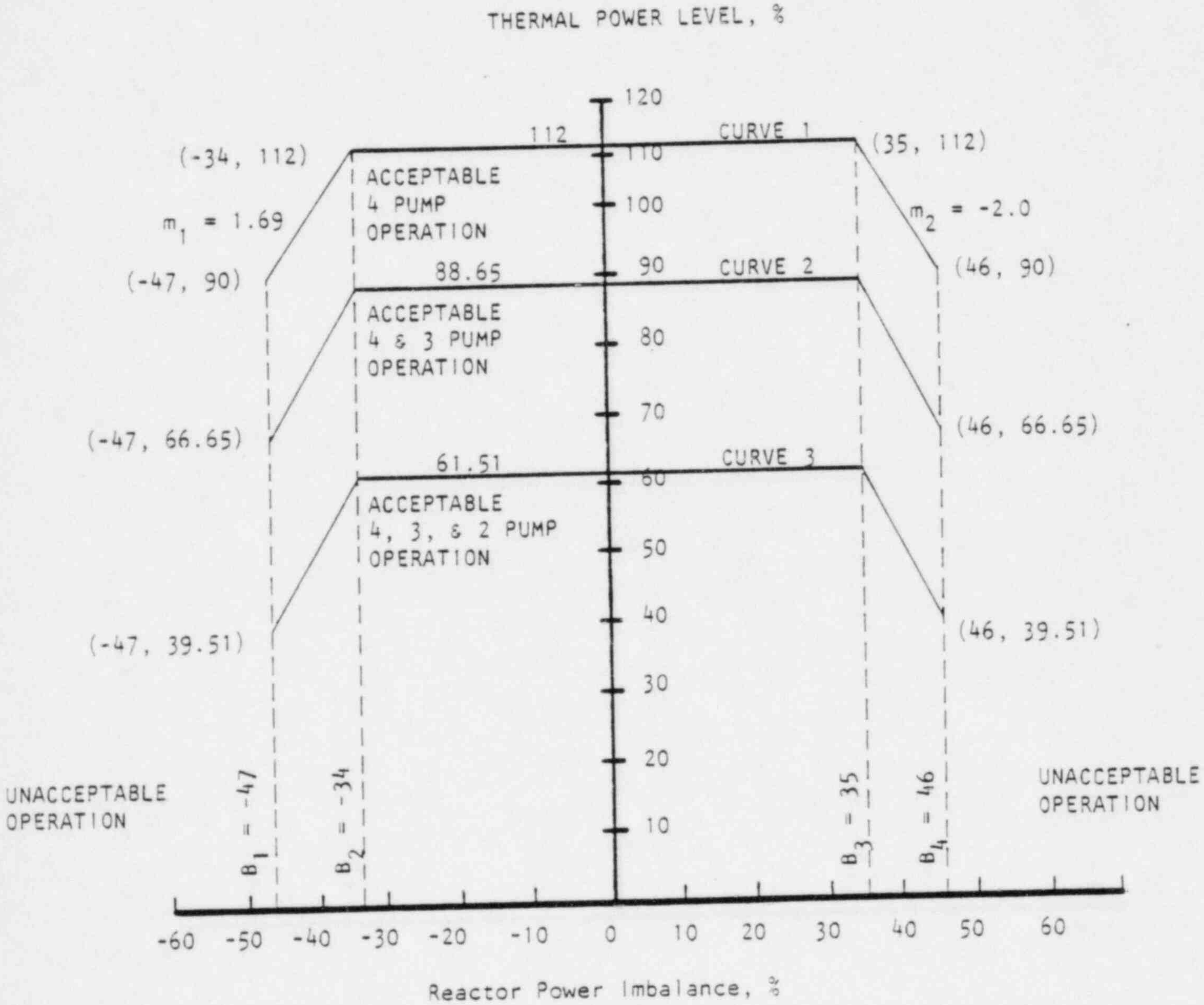


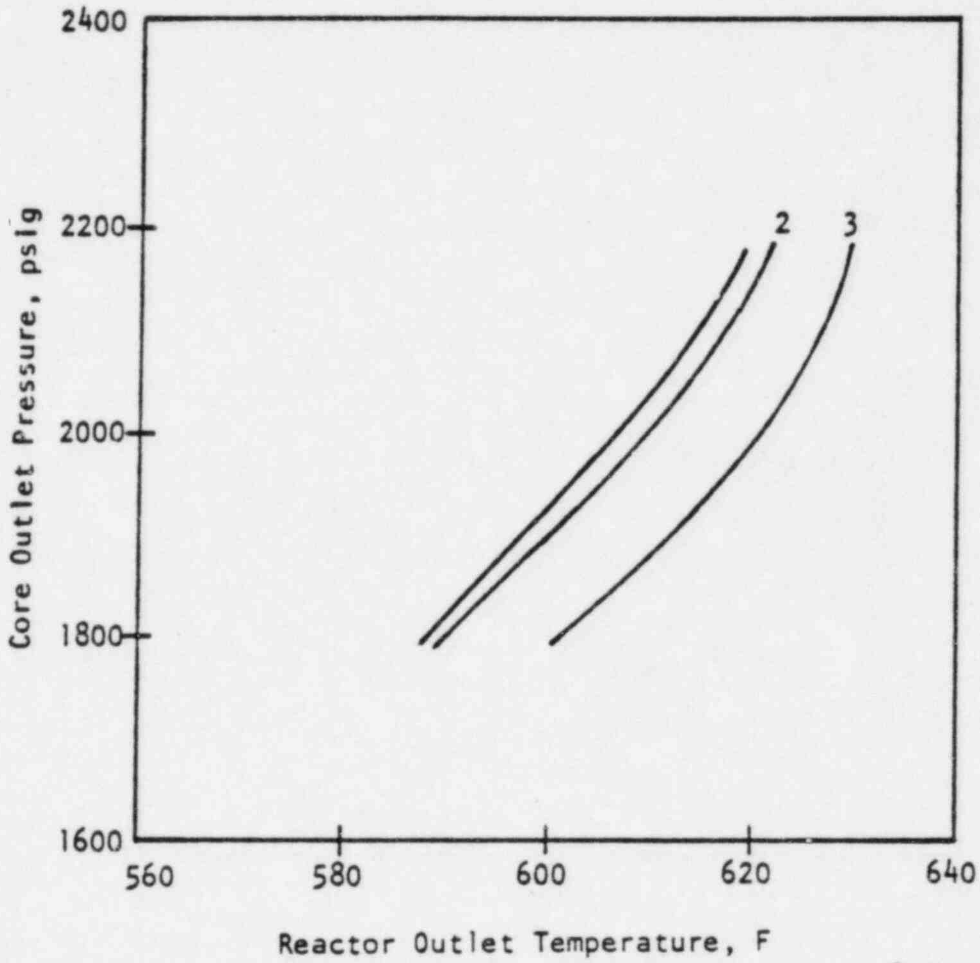
Figure 2.1-2 CORE PROTECTION SAFETY LIMITS, REACTOR POWER IMBALANCE (CYCLE 5)



SMUD CYCLE 5 CORE PROTECTION SAFETY LIMITS

CURVE	REACTOR COOLANT DESIGN FLOW, GPM
1	387,600
2	288,374
3	187,986

FIGURE 2.1-3. Core Protective Safety Bases



Curve	Reactor coolant flow, gpm	Power,%	Pumps operating (type of limit)
1	387600	112	Four (DNBR Limit)
2	288374	88.65	Three (DNBR Limit)
3	187986	61.51	One in each loop (quality limit)

2.2 SAFETY LIMITS, REACTOR SYSTEM PRESSURE

Applicability

Applies to the limit on reactor coolant system pressure.

Objective

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

Specification

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

2.2.2 The nominal setpoint of the pressurizer code safety valves shall be less than or equal to 2500 psig.

Bases

The reactor coolant system (1) serves as a barrier to prevent radionuclides in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure, the reactor coolant system is a barrier against the release of fission products. Establishing a system limit helps to assure the integrity of the reactor coolant system. The maximum transient pressure allowable in the reactor coolant system pressure vessel under the ASME code, Section III, is 110 percent of design pressure. (2) The maximum transient pressure allowable in the reactor coolant system piping, valves, and fittings under ANSI Section B31.7 is 110 percent of design pressure. Thus, the safety limit of 2750 psig (110 percent of the 2500 psig design pressure) has been established. (2) The settings for the reactor high pressure trip (2300 psig) and the pressurizer code safety valves (2500 psig) (3) have been established to assure that the reactor coolant system pressure safety limit is not exceeded. The initial hydrostatic test was conducted at 3125 psig (125 percent of design pressure) to verify the integrity of the reactor coolant system. Additional assurance that the reactor coolant system pressure does not exceed the safety limit is provided by setting the pressurizer electromatic relief valve at 2450 psig. This setpoint is above normal transients limited by setting the reactor trip at $\leq$ 2300 psig and sufficiently low to assure limited dependence on safety valves operation.

REFERENCES

(1) FSAR, section 4

(2) FSAR, paragraph 4.3.8.1

(3) FSAR, paragraph 4.2.4

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Applicability

Applies to instruments monitoring reactor power, reactor power imbalance, reactor coolant system pressure, reactor coolant outlet temperature, flow, number of pumps in operation, and high Reactor Building pressure.

Objective

To provide automatic protection action to prevent any combination of process variables from exceeding a safety limit.

Specification

2.3.1 The reactor protection system trip setting limits and the permissible bypasses for the instrument channels shall be as stated in table 2.3-1 and figure 2.3-2.

Bases

The reactor protection system consists of four instrument channels to monitor each of several selected plant conditions which will cause a reactor trip if any one of these conditions deviates from a pre-selected operating range to the degree that a safety limit may be reached.

The trip setting limits for protection system instrumentation are listed in table 2.3-1. The safety analysis has been based upon these protection system instrumentation trip set points plus calibration and instrumentation errors.

Nuclear Overpower

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

During normal plant operation with all reactor coolant pumps operating, reactor trip is initiated when the reactor power level reaches 104.9 percent of rated power. Adding to this the possible variation in trip set points due to calibration and instrument errors, the maximum actual power at which a trip would be actuated could be 112 percent, which was used in the safety analysis. (4)

A. Overpower trip based on flow and imbalance

The power level trip set point produced by the reactor

2.3 (Continued)

Nuclear Overpower (Continued)

A. (Continued)

coolant system flow is based on a power-to-flow ratio which has been established to accommodate the most severe thermal transient considered in the design, the loss-of-coolant flow accident from high power. The analysis in section 14 demonstrates the adequacy of the specificed power to flow ratio.

The power level trip set point produced by the power-to-flow ratio provides both high power level and low flow protection in the event the reactor power level increases or the reactor coolant flow rate decreases. The power level trip set point produced by the power to flow ratio provides overpower DNB protection for all modes of pump operation. For every flow rate there is a maximum permissible power level, and for every power level there is a minimum permissible low flow rate. Typical power level and low flow rate combinations for the pump situations of table 2.3-1 are as follows:

1. Trip would occur when four reactor coolant pumps are operating if power is 106 percent and reactor flow rate is 100 percent, or flow rate is 94.34 percent and power level is 100 percent.

2. Trip would occur when three reactor coolant pumps are operating if power is 78.8 percent and reactor flow rate is 74.4 percent or flow rate is 70.75 percent and power level is 75 percent.

3. Trip would occur when one reactor coolant pump is operating in each loop (total of two pumps operating) if the power is 51.4 percent and reactor flow rate is 48.5 percent or flow rate is 46.22 percent and the power level is 49 percent.

For safety analysis calculations the maximum calibration and instrumentation errors for the power level were used.

The power-imbalance boundaries are established in order to prevent reactor thermal limits from being exceeded. These thermal limits are either power peaking kW/ft limits or DNBR limits. The reactor power imbalance (power in the top half of core minus power in the bottom half of core) reduces the power level trip produced by the power-to-flow ratio so that the boundaries of figure 2.3-2 are produced. The power-to-flow ratio reduces the power level trip and associated reactor-power reactor-power-imbalance boundaries by 1.06 percent for a 1 percent flow reduction.

2.3 (Continued)

Nuclear Overpower (Continued)

B. Pump Monitors

The pump monitors prevent the minimum core DNBR from decreasing below 1.3 by tripping the reactor due to (a) the loss of two reactor coolant pumps in one reactor coolant loop, and (b) loss of one or two reactor coolant pumps during two-pump operation. The pump monitors also restrict the power level to 55 percent for one reactor coolant pump operation in each loop.

C. Reactor coolant system pressure

During a startup accident from low power or a slow rod withdrawal from high power, the system high pressure trip set point is reached before the nuclear overpower trip set point. The trip setting limit shown in figure 2.3-1 for high reactor coolant system pressure (2300 psig) has been established to maintain the system pressure below the safety limit (2750 psig) for any design transient and minimize the challenges to the EMOV and code safeties. (1)

The low pressure (1900 psig) and variable low pressure ($12.96\ T_{out} - 5834$) trip set point shown in figure 2.3-1 have been established to maintain the DNB ratio greater than or equal to 1.3 for those design accidents that result in a pressure reduction. (2,3)

Due to the calibration and instrumentation errors the safety analysis used a variable low reactor coolant system pressure trip value of ($12.96\ T_{out} - 5884$)

D. Coolant outlet temperature

The high reactor coolant outlet temperature trip setting limit (618 F) shown in figure 2.3-1 has been established to prevent excessive core coolant temperatures in the operating range. Due to calibration and instrumentation errors, the safety analysis used a trip set point of 620 F.

E. Reactor Building pressure

The high Reactor Building pressure trip setting limit (4 psig) provides positive assurance that a reactor trip will occur in the unlikely event of a steam line failure in the Reactor Building or a loss-of-coolant accident, even in the absence of a low pressure trip.

2.3 (Continued)

F. Shutdown bypass

In order to provide for control rod drive tests, zero power physics testing, and startup procedures, there is provision for bypassing certain segments of the reactor protection system. The reactor protection system segments which can be bypassed are shown in Table 2.3-1. Two conditions are imposed when the bypass is used:

1. By administative control the nuclear overpower trip set point must be reduced to a value of $\leq$ 5.0 percent of rated power during reactor shutdown.

2. A high reactor coolant system pressure trip set point of 1820 psig is automatically imposed.

The purpose of the 1820 psig high pressure trip set point is to prevent normal operation with part of the reactor protection system bypassed. This high pressure trip set point is lower than the normal low pressure trip set point so that the reactor must be tripped before the bypass is initiated. The overpower trip set point of $\leq$ 5.0 percent prevents any significant reactor power from being produced when performing the physics tests. Sufficient natural circulation (5) would be available to remove 5.0 percent of rated power if none of the reactor coolant pumps were operating.

REFERENCES

(1) FSAR, paragraph 14.1.2.2

(2) FSAR, paragraph 14.1.2.7

(3) FSAR, paragraph 14.1.2.8

(4) FSAR, paragraph 14.1.2.3

(5) FSAR, paragraph 14.1.2.6

Table 2.3-1

REACTOR PROTECTION SYSTEM TRIP SETTING LIMITS

	Four Reactor Coolant Pumps Operating (Nominal Operating Power - 100%)	Three Reactor Coolant Pumps Operating (Nominal Operating Power - 75%)	One Reactor Coolant Pump Operating in Each Loop (Nominal Operating Power - 49%)	Shutdown Bypass
1. Nuclear Power, % of rated, max.	104.9	104.9	104.9	5.0(3)
2. Nuclear power based on flow(2) and Imbalance, % of rated, max.	1.06 times flow minus reductic~ due to Imbalance(s)	1.06 times flow minus reduction due to Imbalance(s)	1.06 times flow minus reduction due to Imbalance(s)	Bypassed
3. Nuclear power based on pump monitors, % of rated, max.	NA	NA	55	Bypassed
4. High reactor coolant system pressure, psig max.	2300	2300	2300	1820(4)
5. Low reactor coolant system Pressure, psig min.	1900	1900	1900	Bypassed
6. Variable low reactor coolant system pressure, psig min.	12.96 T_{out}-5834	12.96 T_{out}-5834	12.96 T_{out}-5834	Bypassed
7. Reactor coolant temp. F., max.	618	618	618	618
8. High Reactor Building pressure, psig max.	4	4	4	4

(1) T_{out} is in degrees Fahrenheit (F).

(2) Reactor coolant system flow, %.

(3) Administratively controlled reduction set only during reactor shutdown.

(4) Automatically set when other segments of the RPS (as specified) are bypassed.

(5) The pump monitors also produce a trip on: (a) loss of two reactor coolant pumps in one reactor coolant loop, and (b) loss of one or two reactor coolant pumps durng two-pump operation.

Figure 2.3-1 Protective System Maximum Allowable Setpoints, Pressure Vs Temperature

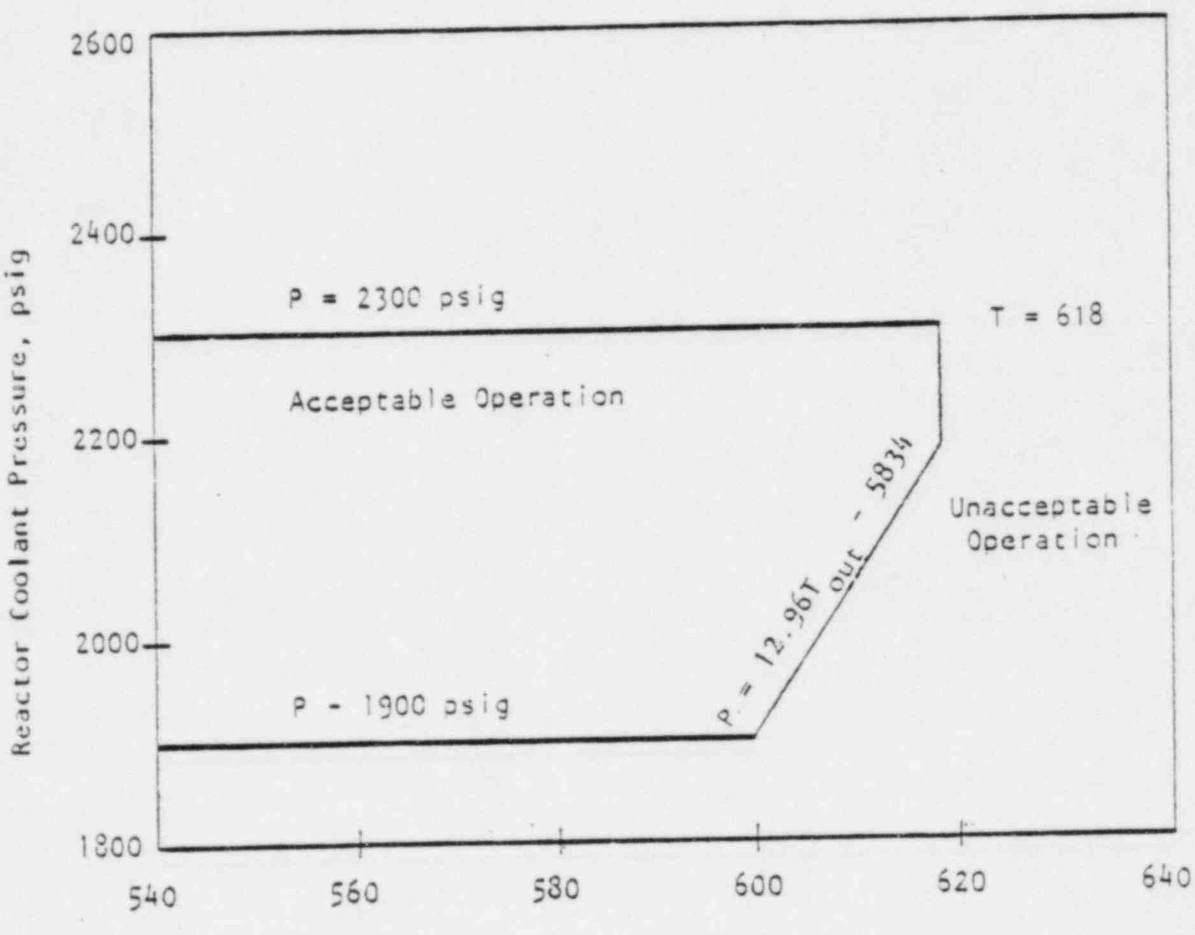
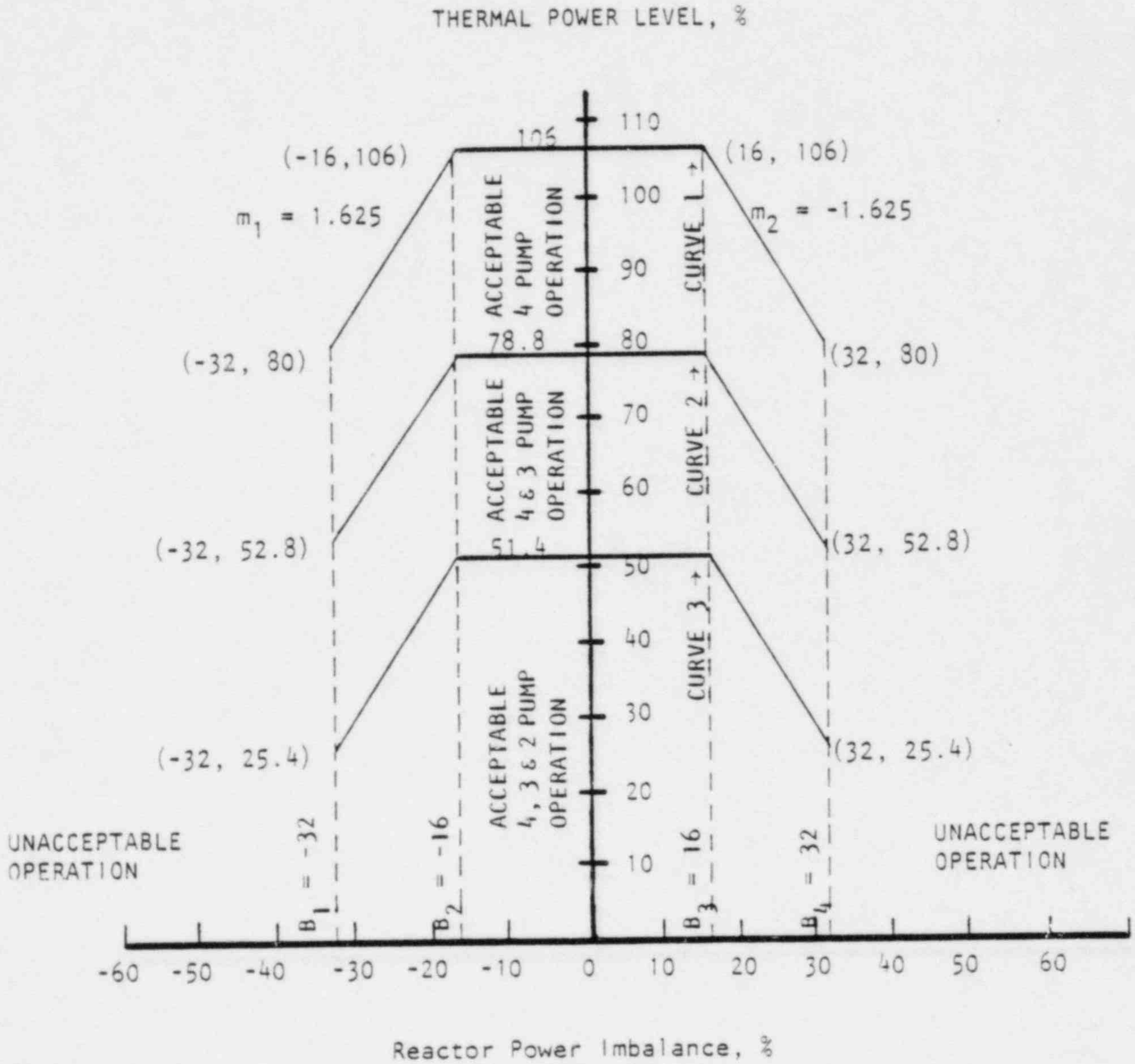


Figure 2.3-2 PROTECTIVE SYSTEM MAXIMUM ALLOWABLE SETPOINTS, REACTOR POWER IMBALANCE (CYCLE 5)



SMUD CYCLE 5 MAXIMUM ALLOWABLE SETPOINTS

CURVE	REACTOR COOLANT DESIGN FLOW, GPM
1	387,600
2	288,374
3	187,986

3. LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the reactor coolant system.

Objective

To specify those limiting conditions for operation of the reactor coolant system which must be met to ensure safe reactor operations.

3.1.1 Operational Components

Specification

3.1.1.1 Reactor Coolant Pumps

A. Pump combinations permissible for given power levels shall be as shown in specification table 2.3-1.

B. The boron concentration in the reactor coolant system shall not be reduced unless at least one reactor coolant pump or one decay heat removal pump is circulating reactor coolant.

C. Operation at power with two pumps shall be limited to 24 hours in any 30 day period.

3.1.1.2 Steam Generator

A. One steam generator shall be operable whenever the reactor coolant average temperature is above 280 F.

3.1.1.3 Pressurizer Safety Valves

A. The reactor shall not remain critical unless both pressurizer code safety valves are operable.

B. When the reactor is subcritical, at least one pressurizer code safety valve shall be operable if all reactor coolant system openings are closed, except for hydrostatic tests in accordance with ASME Boiler and Pressure Vessel Code, Section III.

3.1.1.4 Pressurizer Electromatic Relief Valve

A. The nominal setpoint of the pressurizer electromatic relief valve shall be 2450 psig $\pm$ 10 psig except when required for cold overpressure protection.

3.1 (Continued)

Bases

A reactor coolant pump or decay heat removal pump is required to be in operation before the boron concentration is reduced by dilution with makeup water. Either pump will provide mixing which will prevent sudden positive reactivity changes caused by dilute coolant reaching the reactor. One decay heat removal pump will circulate the equivalent of the reactor coolant system volume in one half hour or less. (1)

The decay heat removal system suction piping is designed for 300 F and 300 psig; thus, the system can remove decay heat when the reactor coolant system is below this temperature. (2) (3)

One pressurizer code safety valve is capable of preventing overpressurization when the reactor is not critical since its relieving capacity is greater than that required by the sum of the available heat sources which are pump energy, pressurizer heaters, and reactor decay heat. (4) Both pressurizer code safety valves are required to be in service prior to criticality to conform to the system design relief capabilities. The code safety valves prevent overpressure for rod withdrawal accidents. (5) The pressurizer code safety valve lift set point shall be set at 2500 psig $\pm$ 1 percent allowance for error and each valve shall be capable of relieving 345,000 lb/h of saturated steam at a pressure not greater than 3 percent above the set pressure.

The electromatic relief valve setpoint was established to prevent operation of the valve during transients.

Two pump operation is limited until further ECCS analysis is performed.

REFERENCES

(1) FSAR Tables 9.5-2, 4.1-1, 4.2-2, 4.2-4, 4.2-5, 4.2-6

(2) FSAR paragraph 9.5.2.2 and 10.2.2

(3) FSAR paragraph 4.2.5

(4) FSAR paragraph 4.3.8.4 and 4.2.4

(5) FSAR paragraph 4.3.6 and 14.1.2.2.3

3.1.2 Pressurization, Heatup, and Cooldown Limitations

Specification

3.1.2.1 Inservice Leak and Hydrostatic Tests:

Pressure temperature limits for the first five EFP years of inservice leak and hydrostatic tests are given in Figure 3.1.2-3. Heatup and cooldown rates shall be restricted according to the rates specified in Figure 3.1.2-3.

3.1.2.2 Heatup Cooldown:

For the first five EFP years of power operation, the reactor coolant pressure and the system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1.2-1 and Figure 3.1.2-2 respectively. Heatup and cooldown rates shall not exceed the rates stated on the associated figure.

3.1.2.3 The secondary side of the steam generator shall not be pressurized above 200 psig if the temperature of the steam generator shell is below 130 F.

3.1.2.4 The pressurizer heatup and cooldown rates shall not exceed 100 F in any 1-hour period.

3.1.2.5 The spray shall not be used if the temperature difference between the pressurizer and spray fluid is greater than 410 F.

3.1.2.6 Prior to exceeding five effective full power years of operation, Figures 3.1.2-1, -2, and -3 shall be updated for the next service period in accordance with 10 CFR 50, Appendix G, Section V.B. The highest predicted adjusted reference temperature of all the beltline materials shall be used to determine the adjusted reference temperature at the end of the service period. The basis for this prediction shall be submitted for NRC staff review in accordance with Specification 3.1.2.7.

3.1.2.7 The updated proposed technical specifications referred to in 3.1.2.6 shall be submitted for NRC review at least 90 days prior to the end of the service period. Appropriate additional NRC review time shall be allowed for proposed technical specifications submitted in accordance with 10 CFR 50, Appendix G, Section V.C.

Specification (Continued)

3.1.2.8 In the emergency/faulted condition when there is no forced or natural circulation in the reactor coolant system and there is high pressure injection and/or makeup addition, the Reactor Coolant System temperature and pressure shall be limited in accordance with the limit line shown on Figure 3.1.2-4. Under the above emergency/faulted conditions, Figure 3.1.2-2 will not apply.

Bases

The pressure-temperature limits of the reactor coolant pressure boundary are established in accordance with the requirements of Appendix G to 10 CFR 50 and with the thermal and loading cycles used for design purposes.

The limitations prevent non-ductile failure during normal operation, including anticipated operational occurences and system hydrostatic test. The limits also prevent exceeding stress limits during cyclic operation. The loading conditions of interest include:

1. Normal heatup

2. Normal cooldown

3. Inservice leak and hydrostatic test

The major components of the reactor coolant pressure boundary have been analyzed in accordance with Appendix G to 10 CFR 50. The closure head region, reactor vessel outlet nozzles and the beltline region have been identified to be the only regions of the reactor vessel, and consequently of the reactor coolant pressure boundary, that determine the pressure-temperature limitations concerning non-ductile failure.

The closure head region is significantly stressed at relatively low temperatures (due to mechanical loads resulting from bolt pre-load). After 5 EFPY of neutron irradiation exposure, the RT NDT temperature of the beltline region materials will be high enough so that the beltline region of the reactor vessel will control much of the pressure-temperature limitations of the reactor coolant pressure boundary. For the service period for which the limit curves are established, the maximum allowable pressure as a function of fluid temperature is obtained through a point-by-point comparison of the limits imposed by the closure head region, outlet nozzles, and beltline region.

3.1.2 (Continued)

Bases (Continued)

The maximum allowable pressure is taken to be the lowest pressure of the three calculated pressures. The pressure limit is adjusted for the pressure differential between the point of system pressure measurement and the limiting component for all reactor coolant pump combinations. The limit curves were prepared based upon the most limiting adjusted reference temperature of all the beltline region materials at the end of the fifth effective full power year.

The actual shift in RT NDT of the beltline region material will be established periodically during operation by removing and evaluating, in accordance with Appendix H to 10 CFR 50, reactor vessel material irradiation surveillance specimens installed near the inside wall of this or a similar reactor vessel in the core area. Because the neutron energy spectra at the specimen location and at the vessel inner wall location are essentially the same, the measured transition shift for a sample can be applied with confidence to the
adjacent section of the reactor vessel. The limit curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

The unirradiated impact properties of the beltline region materials, required by Appendices G and H to 10 CFR 50, were determinded for those materials for which sufficient amounts of material were available. The adjusted reference temperatures are calculated by adding the radiation-induced ΔRT_{NDT} and the unirradiated RT_{DNT}. The predicted ΔRT_{NDT} are calculated using the respective neutron fluence and copper and phosphorus contents in accordance with Reg. Guide 1.99.

The assumed RT_{NDT} of the closure head region is 60 F and the outlet nozzle steel forgings is 60 F.

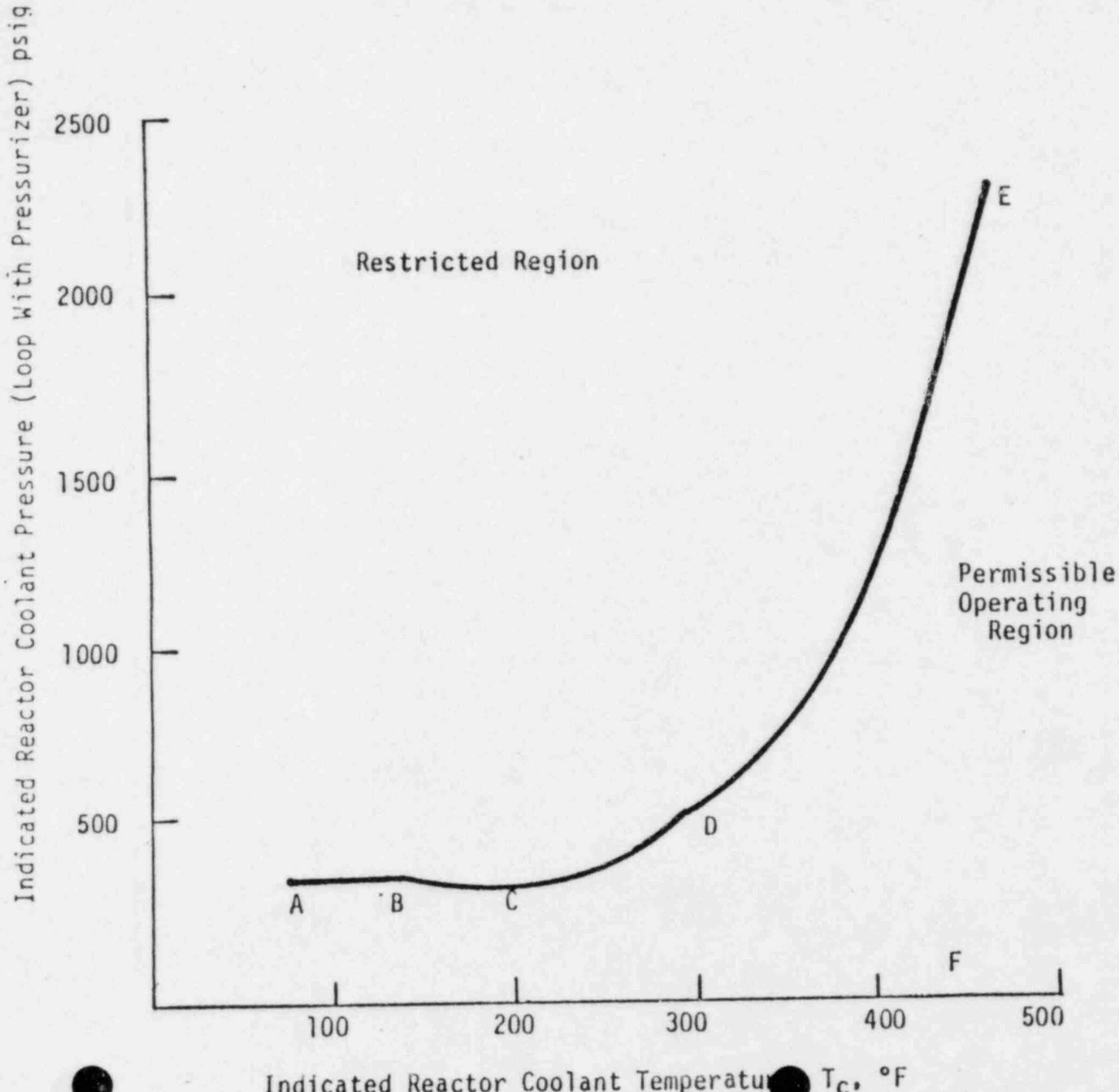
The limitations imposed on pressurizer heatup and cooldown and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME code requirements.

The limitations during emergency/faulted operation when all reactor coolant flow and all feedwater flow is lost to the OTSG's are established to take into consideration that HPI gives false cold leg temperatures. This transient is controlled by Figure 3.1.2-4 and the vessel beltline temperature is calculated using incore thermocouples and subtracting 150 F for conservatism. When the coolant flow or feedwater flow is re-established, a four hour transition period will be allowed to progress from Figure 3.1.2-4 to Figure 3.1.2-2.

FIGURE 3.1.2-1

REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITS

FOR HEATUP FOR THE FIRST 5 EFPY



MAXIMUM HEATUP RATES

TEMP	MAX RATE
70°F-100°F	5°F/HR
100°F-135°F	25°F/HR
135°F-185°F	50°F/HR
185°F-550°F	75°F/HR

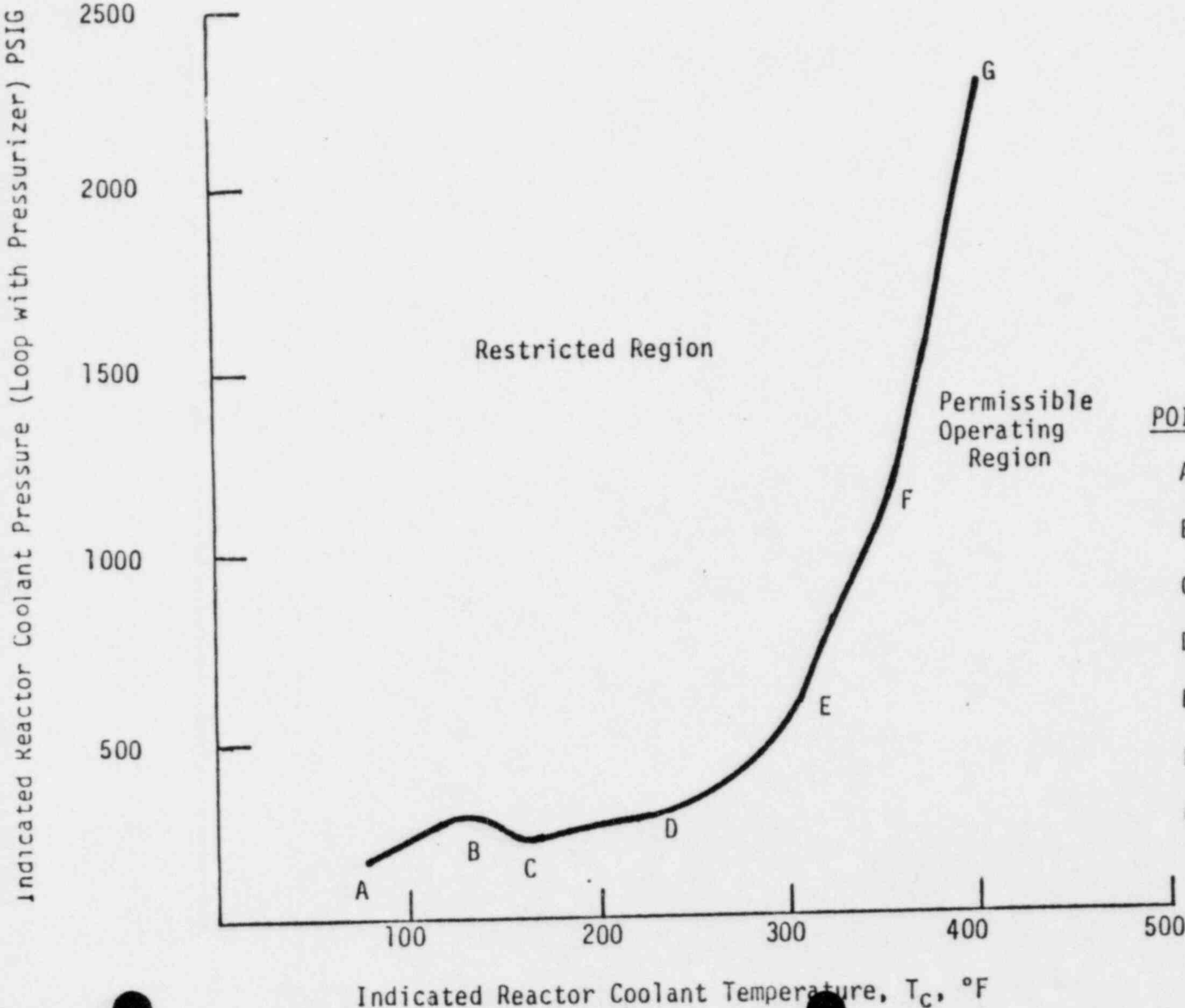
POINT	PRESS	TEMP
A	336	70
B	336	122
C	310	175
D	525	295
E	2250	458

MARGINS OF 25 PSIG 10°F ARE INCLUDED FOR POSSIBLE INSTRUMENTATION ERROR

FIGURE 3.1.2-2

REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITS

FOR COOLDOWN FOR THE FIRST 5 EFPY



COOLDOWN RATE

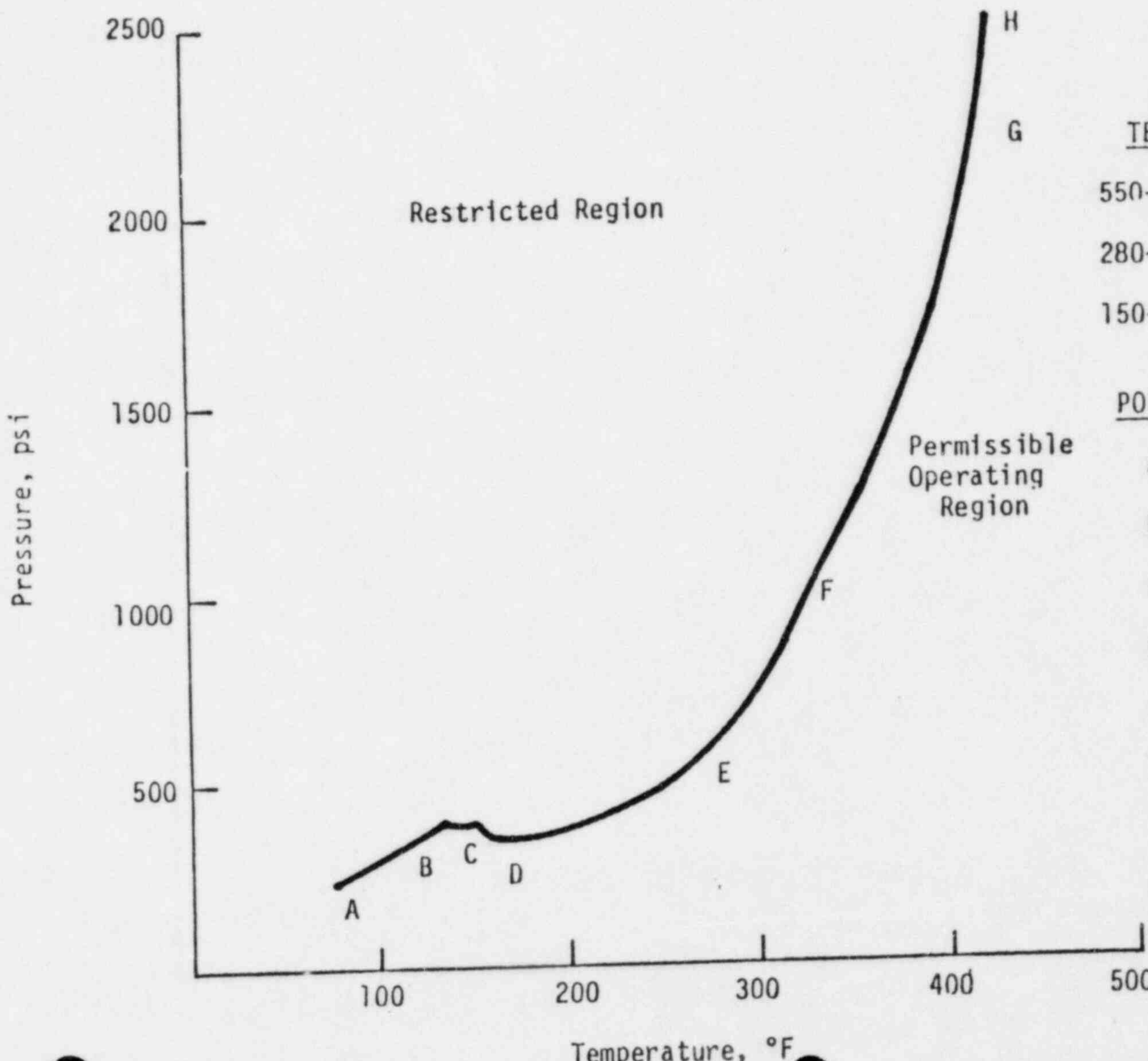
TEMP	MAX C.D. RATE
550-280	100°/HR
280-150	50°/HR
150-	25°/HR

POINT	PRESS	TEMP
A	150	80°
B	275	125°
C	225	163°
D	315	250°
E	525	300°
F	1250	362°
G	2250	407°

MARGINS OF 25 PSIG and 10°F ARE INCLUDED FOR POSSIBLE INSTRUMENTATION ERROR

FIGURE 3.1.2-3

INSERVICE LEAK AND HYDROSTATIC TEST (5 EFPY) HEATUP AND COOLDOWN



MAX HEATUP RATE ≤ 50°F/HR

MAX COOLDOWN RATES:

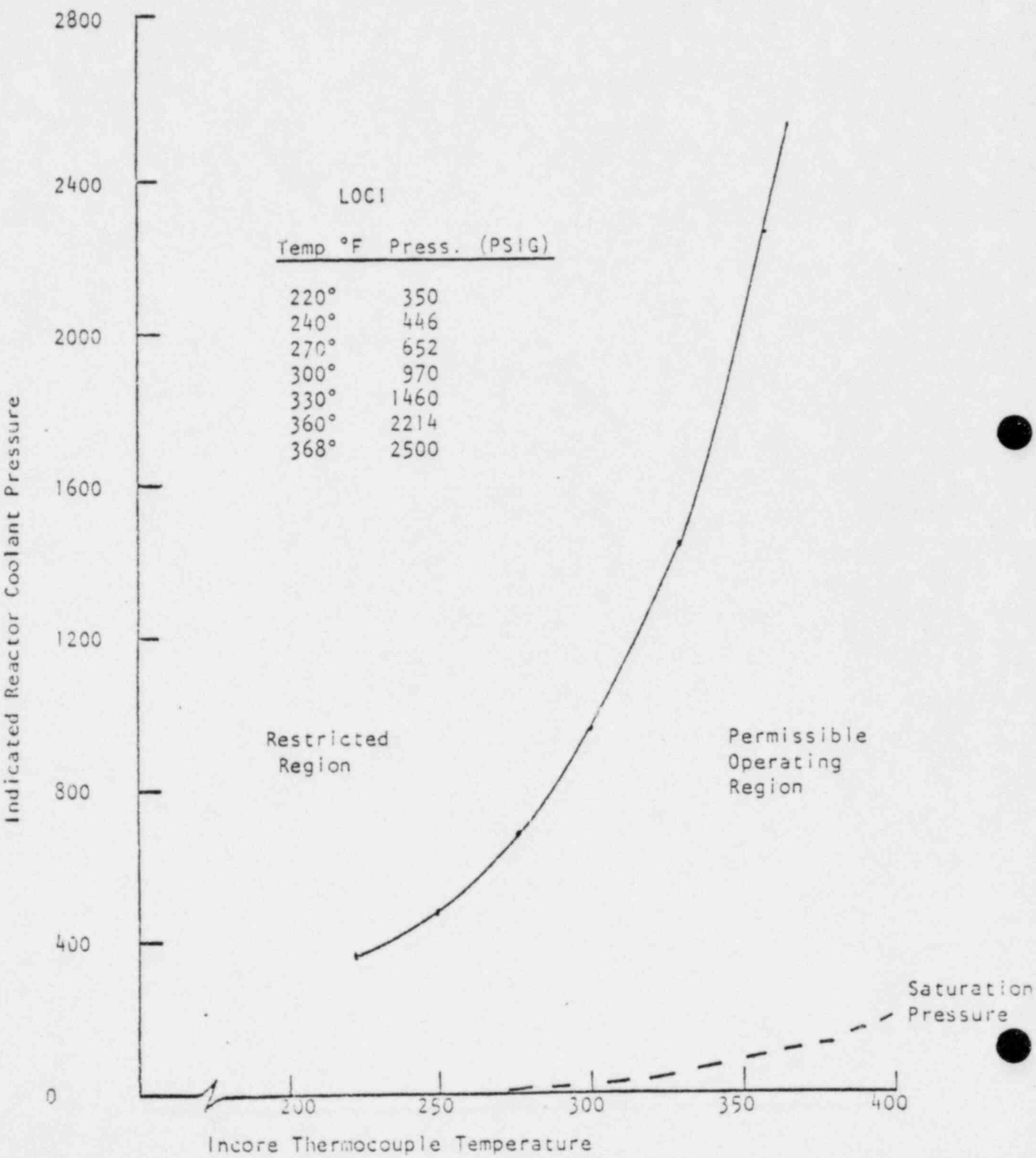
TEMP	C.D. MAX RATES
550-280	100°F/HR
280-150	50°F/HR
150-	25°F/HR

POINT	PRESS	TEMP
A	230	75
B	380	132
C	370	144
D	330	165
E	575	275
F	950	322
G	2250	419
H	2500	426

MARGINS OF 25 PSIG and 10°F ARE INCLUDED FOR POSSIBLE INSTRUMENTATION ERROR

Figure 3.1.2-4

REACTOR COOLANT SYSTEM, EMERGENCY/FAULTED CONDITION-COOLDOWN
LIMITATIONS, APPLICABLE FOR 5.0 EFFECTIVE FULL POWER YEARS



Limiting Conditions for Operation

3.1.3 Minimum Conditions for Criticality

Specifications

3.1.3.1 The reactor coolant temperature shall be above 525 F except for portions of low power physics testing when the requirements of Specification 3.1.8 shall apply.

3.1.3.2 Reactor coolant temperature shall be above DTT + 10 F.

3.1.3.3 When the reactor coolant temperature is below the minimum temperature specified in 3.1.3.1 above, except for portions of low power physics testing when the requirements of Specification 3.1.8 shall apply, the reactor shall be subcritical by an amount equal to or greater than the calculated reactivity insertion due to depressurization.

3.1.3.4 The reactor shall be maintained subcritical by at least 1 percent $\Delta k/k$ until a steam bubble is formed and an indicated water level between 10 and 316 inches is established in the pressurizer.

3.1.3.5 Except for physics tests and as limited by 3.5.2.1 and 3.5.2.5, safety rod groups shall be fully withdrawn prior to any other reduction in shutdown margin by deboration or regulating rod withdrawal during the approach to criticality. Following safety rod withdrawal, the regulating rods shall be positioned within their position limits as defined by specification 3.5.2.5 prior to deboration.

At the beginning of life of the initial fuel cycle, the moderator temperature coefficient is expected to be slightly positive at operating temperatures with the operating configuration of control rods. (1) Calculations show that above 525 F the positive moderator coefficient is acceptable.

Since the moderator temperature coefficient at lower temperatures will be less negative or more positive than at operating temperature, (2) startup and operation of the reactor when reactor coolant temperature is less than 525 F is prohibited except where necessary for low power physics tests.

The potential reactivity insertion due to the moderator pressure coefficient (2) that could result from depressurizing the coolant from 2185 psia to saturation pressure of 885 psia is approximately 0.1 percent $\Delta k/k$.

During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient (1) and the small integrated $\Delta k/k$ would limit the magnitude of a power excursion resulting from a reduction of moderator density.

Limiting Conditions for Operation

3.1.3 (Continued)

Bases (Continued)

The requirement that the reactor is not to be made critical below DTT + 10 F provides increased assurances that the proper relationship between primary coolant pressure and temperatures will be maintained relative to the NDTT of the primary coolant system. Heatup to this temperature will be accomplished by operating the reactor coolant pumps. The DTT at BOL for the most limiting component in the reactor coolant system is less than +100 F.

If the shutdown margin required by Specification 3.5.2 is maintained, there is no possibility of an accidental criticality as a result of a decrease of coolant pressure.

The requirement for pressurizer bubble formation and specified water level when the reactor is less than 1 percent subcritical will assure that the reactor coolant system cannot become solid in the event of a rod withdrawal accident or a start-up accident and that the water level is above the minimum detectable level.

The requirement that the safety rod groups be fully withdrawn before criticality ensures shutdown capability during startup. This does not prohibit rod latch confirmation, i.e., withdrawal by group to a maximum of 3 inches withdrawn of all seven groups prior to safety rod withdrawal.

REFERENCES

(1) FSAR, section 3

(2) FSAR, paragraph 3.2.1.4

3.1.4 Reactor Coolant System Activity

Specification

3.1.4.1 The total fission product activity of the reactor coolant due to nuclides with half lives longer than 30 minutes shall not exceed $43/\bar{E}$ microcuries per gm whenever the reactor is critical. $\bar{E}$ is the average (mean) beta and gamma energies per disintegration, in MeV, weighted in proportion to the measured activity of the radionuclides in reactor coolant samples.

Bases

The above specification is based on limiting the consequences of a postulated accident involving the double-ended rupture of a steam generator tube. The rupture of a steam generator tube enables reactor coolant and its associated activity to enter the secondary system where volatile isotopes could be discharged to the atmosphere through condenser air-ejectors and through steam safety valves (which may lift momentarily). Since the major portion of the activity entering the secondary system is due to noble gases, the bulk of the activity would be discharged to the atmosphere. The activity release continues until the operator stops the leakage by reducing the reactor coolant system pressure below the set point of the steam safety valves and isolates the faulty steam generator. The operator can identify a faulty steam generator by using the off-gas monitors on the condenser air ejector lines; thus he can isolate the faulty steam generator within 34 minutes after the tube break occurred. During that 34 minute period, a maximum of 2740 ft^3 of hot reactor coolant leaked into the secondary system; this is equivalent to a cold volume of 1980 ft^3.

The controlling dose for the steam generator tube rupture accident is the whole-body dose resulting from immersion in the cloud of released activity. To insure that the public is adequately protected, the specific activity of the reactor coolant will be limited to a value which will insure that the whole-body dose at the site boundary will not exceed 0.5 rem, the limit in 10 CFR Part 20 for whole body dose in an unrestricted area.

Although only volatile isotopes will be released from the secondary system, the following whole-body dose calculation conservatively assumes that all of the radioactivity which enters the secondary system with the reactor coolant is released to the atmosphere. Both the beta and gamma radiation from these isotopes contribute to the whole-body dose. 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

3.1.4 (Continued)

Bases (Continued)

assumed that meteorological conditions during the course of the accident correspond to Pasquil Type F and 0.6 meter per second wind speed, resulting in a X/Q value of 8.51×10^{-4} sec/m^3.

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

$$\text{Dose (Rem)} = 0.246 \cdot \bar{E} \cdot A \cdot V \cdot X/Q \cdot \rho$$

$$A_{max}\ (\text{uCi/gm}) = \frac{(\text{Dose})_{max}}{0.246 \cdot \bar{E} \cdot V \cdot X/Q \cdot \rho} = \frac{0.5}{0.246 \times \bar{E} \times 77.6 \times 8.51 \times 10^{-4} \times 0.713}$$

$$A_{max}\ (\text{uCi/gm}) = 43/\bar{E}$$

Where

A = Reactor coolant activity (uCi/ml = Ci/m^3)

V = Volume of hot reactor coolant leaked into secondary system (2740 ft^3 = 77.6 m^3)

X/Q = Atmospheric dispersion coefficient at site boundary for a two hour period (8.51×10^{-4} sec/m^3)

$\bar{E}$ = Average beta and gamma energies per disintegration (MeV)

ρ = Density of hot reactor coolant (0.713 gm/cc)

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

A. Quantitative measurement of the specific activity (in units of uCi/gm) of radionuclides with half lives longer than 30 minutes, which make up at least 95 percent of the total activity in reactor coolant samples.

B. A determination of the average beta and gamma decay energies per disintegration for each nuclide, measured in (A) above, by utilizing known decay energies and decay schemes (e.g., Table of Isotopes, Sixth Edition, March 1968).

C. A calculation of $\bar{E}$ by the average beta and gamma energy for each radionuclide in proportion to its specific activity, as measured in (A) above.

3.1.5 Chemistry

Applicability

Applies to the limiting conditions of reactor coolant chemistry for continous operation of the reactor.

Objective

To protect the reactor coolant system from the effects of impurities in the reactor coolant.

Specification

3.1.5.1 The following limits shall not be exceeded for the listed reactor coolant conditions.

Contaminant	Specification	Reactor Coolant Conditions
Oxygen as O_2	0.10 ppm max	above 250 F
Chloride as Cl^-	0.15 ppm max	above cold shutdown conditions
Fluoride as F^-	0.15 ppm max	above cold shutdown conditions

3.1.5.2 During operation above 250 F, if any of the specifications in 3.1.5.1 is exceeded, corrective action shall be initiated within 8 hours. If the concentration limit is not restored within 24 hours after initiation of corrective action, the reactor shall be placed in a cold shutdown condition using normal procedures.

3.1.5.3 During operations between 250 F and cold shutdown conditions, if the chloride or fluoride specification in 3.1.5.1 are exceeded, corrective action shall be initiated within 8 hours to restore the normal operating limits. If the specifications are not restored within 24 hours after initiation of corrective action, the reactor shall be placed in a cold shutdown condition using normal procedures.

3.1.5.4 If the oxygen concentration and either the chloride or fluoride concentration of the primary coolant system exceed 1.0 ppm the reactor shall be immediately brought to the hot shutdown condition using normal shutdown procedures, and action is to be taken immediately to return the system to within normal operation specifications. If specifications given in 3.1.5.1 have not been reached in 12 hours, the reactor shall be brought to a cold shutdown condition using normal procedures.

3.1.5 (Continued)

Bases

By maintaining the chloride, fluoride, and oxygen concentration in the reactor coolant within the specifications, the integrity of the reactor coolant system is protected against potential stress corrosion attack. (1,2).

The oxygen concentration in the reactor coolant system is normally expected to be below detectable limits since dissolved hydrogen is used when the reactor is critical and a residual of hydrazine is used when the reactor is subcritical to control the oxygen. The requirement that the oxygen concentration not exceed 0.1 ppm is added assurance that stress corrosion cracking will not occur. (3).

If the oxygen, chloride, or fluoride limits are exceeded, measures can be taken to correct the condition (e.g., switch to the spare demineralizer, replace the ion exchange resin, increase the hydrogen concentration in the makeup tank, etc.) and further because of the time dependent nature of any adverse effects arising from halogen or oxygen concentrations in excess of the limits, it is unnecessary to shutdown immediately.

The oxygen and halogen limits specified are at least an order of magnitude below concentrations which could result in damage to materials found in the reactor coolant system even if maintained for an extended period of time. (3) Thus, the period of eight hours to initiate corrective action and the period of 24 hours thereafter to perform corrective action to restore the concentration within the limits have been established. The eight hour period to initiate corrective action allows time to ascertain that the chemical analyses are correct and to locate the source of contamination. If corrective action has not been effective at the end of 24 hours, then the reactor coolant system will be brought to the cold shutdown condition using normal procedures and corrective action will continue.

The maximum limit of 1 ppm for the oxygen and halogen concentration that will not be exceeded was selected as the hot shutdown limit because these values have been shown to be safe at 500 F. (4)

References

(1) FSAR Section 4.1.2.7
(2) FSAR Section 9.2.2
(3) Corrosion and Wear Handbook, O.J. DePaul, Editor
(4) Stress Corrosion of Metals, Logan.

3.1.6 Leakage

Specification

3.1.6.1 If the total reactor coolant leakage rate exceeds 10 gpm, the reactor shall be shutdown within 24 hours of detection, unless leakage has been reduced to less than 10 gpm.

3.1.6.2 If unidentified reactor coolant leakage exceeds 1 gpm or if any reactor coolant leakage is evaluated as unsafe, the reactor shall be shutdown within 24 hours of detection.

3.1.6.3 If any reactor coolant leakage exists through a non-isolable fault in a RCS strength boundary, except steam generator tubes (such as the reactor vessel, piping, valve body, etc.), the reactor shall not remain critical and cooldown to the cold shutdown condition shall be initiated within 24 hours of detection.

3.1.6.4 If reactor shutdown is required by Specification paragraphs 3.1.6.1, 3.1.6.2 or 3.1.6.3, the rate of shutdown and the condition of shutdown shall be determined by the safety evaluation for each case and justified in writing as soon thereafter as practicable.

3.1.6.5 Action to evaluate the safety implication of reactor coolant leakage shall be initiated. The nature, as well as the magnitude, of the leak shall be considered in this evaluation. The safety evaluation shall assure that the exposure of off-site personnel to radiation is within the guidelines of 10 CFR 20.

3.1.6.6 If reactor shutdown is required per Specification paragraphs 3.1.6.1, 3.1.6.2 or 3.1.6.3, the reactor shall not be restarted until the leak is repaired or until the problem is otherwise corrected.

3.1.6.7 During power operation, two reactor coolant leak detection systems of different operating principles shall be in operation, with one of the two systems sensitive to radioactivity. The systems sensitive to radioactivity may be out-of-service for 48 hours provided two other means are available to detect leakage.

3.1.6.8 Indicated leakage of reactor coolant shall be considered actual leakage unless (1) it can be proven that there is no actual leakage or (2) a safety problem does not exist. Loss of reactor coolant through reactor coolant pump seals and system valves to connecting systems which vent to the gas vent header and from which coolant can be returned to the reactor shall not be subject to the considerations of Specification paragraphs 3.1.6.1 - 3.1.6.6 except that such losses when added to leakage shall not exceed 30 gpm.

3.1.6.9 Primary-to-secondary leakage through the steam generator tubes shall be limited to 1 GPM total for all steam generators. With any steam generator tube leakage greater than 1 GPM, reduce leakage to less than 1 GPM; or bring the reactor to cold shutdown conditions within 48 hours.

3.1.6.10 If reactor shutdown is required due to Section 3.1.6.9, restore the inoperable generator(s) to operable status by plugging the leaking tubes prior to increasing the average temperature above 200 F.

Bases

Every reasonable effort will be made to reduce reactor coolant leakage including evaporative losses (which may be on the order of 0.5 gpm) to the lowest possible rate and at least below 1 gpm in order to prevent a large leak from masking the presence of a smaller leak. Evaporative losses identified during startup testing of 0.5 gpm are not considered part of the 1 gpm unidentified leakage. Water inventory balances, radiation monitoring equipment, boric acid crystalline deposits, and physical inspections can disclose reactor coolant leaks. Any leak of radioactive fluid, whether from the reactor coolant system primary boundary or not can be a serious problem with respect to in-plant radioactivity contamination and cleanup or it could develop into a still more serious problem; therefore, first indications of such leakage will be followed up as soon as practicable.

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

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

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

3.1.6 (Continued)

Bases (continued)

A. Sump Levels - All Reactor Building leakage is collected in the Reactor Building sumps. These sumps drain by gravity into a 120 gallon Reactor Building drain accumulation tank. The drain accumulation tank is used to measure the drain flow with level indicators at 20 gallons and 120 gallons. The tank is dumped into the East decay heat removal pump room sump. The frequency of dumping the accumulation tank and time interval between levels are recorded in the Control Room and are direct measures of the flow rate. Depending on the level at which the tank is dumped, the time to confirm a 1 gpm leak is between 40 minutes and 120 minutes.

Frequency of operation of the East DHR pump room sump pumps is recorded in the control room to provide verification of proper operation of the Reactor Building drain accumulation tank.

Since the Reactor Building drain system collects drainage from all components in the Reactor Building, a change in drain flow does not necessarily indicate a reactor coolant system leak. One method available for determining if the additional drain flow is reactor coolant is to collect drainage in the drain accumulation tank, draw a sample from the tank, and analyze the sample for boric acid concentration and radioactivity.

B. Radioactivity - Changes in the reactor coolant leakage rate in the Reactor Building may cause changes in the control room indication of the Reactor Building atmosphere particulate and gas radioactivities and of the Reactor Building radiation monitors.

The response time for the radiation monitors to detect a given leak rate are dependent on the coolant activity level and the minimum detector sensitivity.

For a leak rate of 1 gpm, the following gaseous radiation monitor response times were calculated:

Coolant Activity	Response Time
1% defective fuel	67 seconds
0.1% defective fuel	5.3 minutes

The airborne particulate radiation monitor response time is dependent upon the speed of filter paper advance which, during normal operation, will be the slow speed. Thus, assuming either 0.1 percent defective fuel and 1 gpm leak or expected corrosion product activity and a 1 gpm leak, the response time will be about

3.1.6 (Continued)

Bases (Continued)

B. (Continued)
1 to 2 hours. This time period is associated with filter tape movement from the point of particle deposition to the detector. If leakage is indicated by another leak detection method, the filter paper can be manually advanced to verify that a substantial leak has occurred. By stopping the filter tape advance mechanism, an integrated sample can be taken over a short period of time (e.g. 5 minutes) for a quick evaluation of the situation.

Coolant Activity	Response Time Excluding Filter Advance
1% defective fuel	40 seconds
0.1% defective fuel	41 seconds
No defective fuel, corrosion products only	18 minutes

C. Reactor Coolant Inventory - Total reactor coolant system leakage rate is periodically determined by comparing indications of reactor power, coolant temperature, pressurizer water level and makeup tank level over a time interval. All of these indications are recorded. Since the pressurizer level is maintained essentially constant by the pressurizer level controller, any coolant leakage is replaced by coolant from the makeup tank resulting in a tank level decrease. The makeup tank capacity is 31 gallons per inch of height and each graduation on the level recorder represents 1 inch of tank height. This inventory monitoring method is capable of detecting changes on the order of 31 gallons. A 1 gpm leak would therefore be detectable within approximately one-half hour.

As described above, in addition to direct observation, the means of detecting reactor coolant leakage are based on two different principles, i.e., activity and sump level and reactor coolant inventory measurements. Two systems of different principles provide, therefore, diversified ways of detecting leakage to the reactor building.

The upper limit of 30 gpm is based on the contingency of a complete loss of plant power. A 30 gpm loss of water in conjunction with a complete loss of plant power and subsequent cooldown of the reactor coolant system by the turbine bypass system (set at 1,040 psia) and steam driven emergency

3.1.6 (Continued)

Bases (Continued)

feedwater pump would require more than 60 minutes to empty the pressurizer from the combined effect of system leakage and contraction. This will be ample time to restore electical power to the plant and makeup flow to the reactor coolant system.

The plant is expected to be operated in a manner such that the secondary coolant will be normally maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is consistantly not maintained within these chemistry limits, over some period of time localized corrosion could occur and might result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 1 GPM). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 1 GPM can be detected by monitoring the secondary coolant. Leakage in excess of this limit will require plant shutdown during which the leaking tubes will be located and plugged.

3.1.7 Moderator Temperature Coefficient of Reactivity

Specification

The moderator temperature coefficient shall not be positive at power levels above 95 percent of rated power.

Bases

A non-positive moderator coefficient at power levels above 95 percent of rated power is specified such that the maximum clad temperatures will not exceed the Final Acceptance Criteria based on LOCA analyses. Below 95 percent of rated power the Final Acceptance Criteria will not be exceeded with a positive moderator temperature coefficient of $+0.9 \times 10^{-4}$ $\Delta k/k/F$ corrected to 95 percent of rated power. All other accident analyses as reported in the FSAR have been performed for a range of moderator temperature coefficients including $+0.9 \times 10^{-4}$ $\Delta k/k/F$.

The experimental value of the moderator coefficient will be corrected to obtain the hot full power moderator coefficient. When the hot zero-power value is corrected to obtain the 95 percent power value, the following corrections will be applied:

1. Uncertainty in isothermal measurement - The measured moderator temperature coefficeint will contain uncertainty owing to the ΔT of the base and perturbed conditions and the uncertainty in the reactivity measurement.

   Proper corrections will be added for these conditions to provide a conservative moderator coefficient.

2. Doppler contribution at hot zero power - During measurement of the isothermal moderator coefficient at hot zero power, the fuel temperature will increase by the same amount as the moderator. The measured temperature coefficient must therefore be increased to obtain a pure moderator temperature coefficient.

3. Moderator temperature change - The hot zero power measurement must be corrected for the difference in water temperature at zero power (532 F) and 15 percent power (582 F). Above this power, the average moderator temperature remains 582 F.

4. Fuel temperature interaction (power effect) - The moderator coefficient must be adjusted to account for the interaction of an average moderator temperature with increasing fuel temperatures as power increases. Adjust the moderator coefficient at 15 percent power to the coefficient at any power level above 15 percent.

3.1.7 (Continued)

Bases (Continued)

5. Dissolved boron concentration - This correction is for any difference in boron concentration between zero and full power. Since the moderator coefficient is more positive for greater amounts of dissolved boron, the sign of the correction depends on whether boron is added or removed.

6. Control rod insertion - This correction is for the differences in moderator coefficients between an unrodded and rodded core.

7. Isothermal to distributed temperatures - The correction for spatially distributed moderator temperature effects has been found to be insignificant. Therefore, correction for distributed effects is not required.

REFERENCES

(1) FSAR, subsections 14.1 and 14.2

(2) FSAR, paragraph 3.2.2.1.5.D

3.1.8 Low Power Physics Testing Restrictions

Specification

The following special limitations are placed on low power physics testing.

3.1.8.1 Reactor Protective System Requirements

A. Below 1820 psig shutdown bypass trip setting limits shall apply in accordance with table 2.3-1.

B. Above 1900 psig nuclear overpower trip shall be set at a maximum of 5.0 percent.

3.1.8.2 Startup rate rod withdrawal hold shall be in effect at all times.

3.1.8.3 During low power physics testing, the minimum reactor coolant temperature for criticality shall be 240 F. A minimum shutdown margin of 1 percent $\Delta k/k$ shall be maintained with the highest worth control rod fully withdrawn.

Bases

The above specification provides additional safety margins during low power physics testing. The startup rate rod withdrawal hold is described in paragraph 7.2.2.1.3 and applies to the source and intermediate power ranges.

3.1.9 Control Rod Operation

Specification

3.1.9.1 The concentration of dissolved gases in the reactor coolant shall be limited to 100 std. cc/kilogram of water at the reactor vessel outlet temperature.

3.1.9.2 Allowable combinations of pressure and temperature for control rod operation shall be to the left of and above the limiting pressure versus temperature curve for a dissolved gas concentration of 100 std. cc/kilogram of water as shown in figure 3.1.9-1.

3.1.9.3 In the event the limits of Specifications 3.1.9.1 or 3.1.9.2 are exceeded, the center control rod drive mechanism shall be checked for accumulation of undissolved gases.

Bases

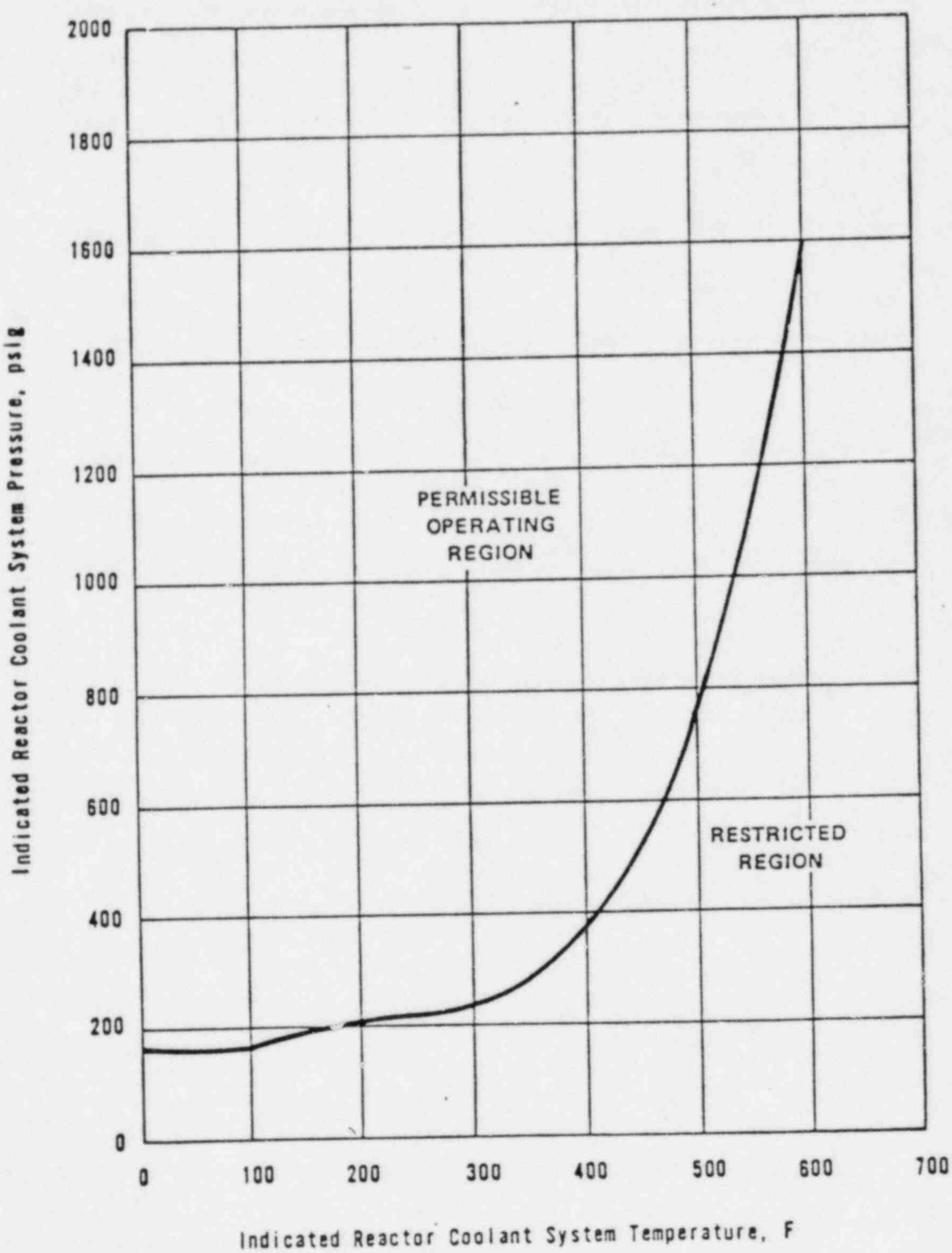
By maintaining the reactor coolant temperature and pressure as specified above, any dissolved gases in the reactor coolant system are maintained in solution.

Although the dissolved gas concentration is expected to be approximately 20-40 std. cc/kilogram of water, the dissolved gas concentration is conservatively assumed to be 100 std. cc/kilogram of water at the reactor vessel outlet temperature.

The limiting pressure versus temperature curve for dissolved gases is determined by the equilibrium pressure versus temperature curve for the dissolved gas concentration of 100 std. cc/kilogram of water. The equilibrium total pressure is the sum of the partial pressure of the dissolved gases plus the partial pressure of water at a given temperature. The margin of error consists of the maximum pressure difference between the pressure sensing tap and lowest pressure point in the system, the maximum pressure gage error, and the pressure difference due to the maximum temperature gage error.

If either the maximum dissolved gas concentration (100 std. cc/kilogram of water) is exceeded or the operating pressure falls below the limiting pressure versus temperature curve, the center CRDM should be checked for accumulation of undissolved gases.

FIGURE 3.1.9-1
LIMITING PRESSURE VERSUS TEMPERATURE
FOR CONTROL ROD DRIVE OPERATION



3.2 HIGH PRESSURE INJECTION AND THE CHEMICAL ADDITION SYSTEMS

Applicability

Applies to the operational status of high pressure injection and the chemical addition systems.

Objective

To provide for adequate boration under all operating conditions to assure ability to bring the reactor to a cold shutdown condition.

Specification

The reactor shall not remain critical unless the following conditions are met:

3.2.1 Two pumps capable of supplying high pressure injection are operable. (also see Specification 3.3.2).

3.2.2 The borated water storage tank and its flow path to the reactor for high pressure injection is operable.

3.2.3 A source of concentrated boric acid solution in addition to the borated water storage tank is available and operable. This requirement is fulfilled by the concentrated boric acid storage tank. This tank shall contain at least the equivalent of 10,000 gallons of 7,100 ppm boron. System piping and valves necessary to establish a flow path for high pressure injection shall also be operable and shall have at least the same temperature as the boric acid storage tank. One associated boric acid pump is operable. The concentrated boric acid storage tank water shall not be less than 70 F, and at least one channel of heat tracing shall be operable for this tank's associated piping. The concentrated boric acid storage tank boron concentration shall not exceed 8,500 ppm boron.

Bases

The makeup and purification system and chemical addition systems provide control of the reactor coolant system boron concentration.(1) This is normally accomplished by using either the makeup pump or one of the two high pressure injection pumps in series with a boric acid pump associated with the concentrated boric acid storage tank. The alternate method of boration will be the use of the makeup of high pressure injection pumps taking suction directly from the borated water storage tank.(2)

3.2 (Continued)

Bases (Continued)

The quantity of boric acid in storage from either of the two above mentioned sources is sufficient to borate the reactor coolant system to a 1 percent subcritical margin in the cold condition (70 F) at the worst time in core life with a stuck control rod assembly. The maximum required is the equivalent of 9,105 gallons of 7100 ppm boron. This requirement is satisfied by requiring of minimum of 10,000 gallons of 7100 ppm in the concentrated borated acid storage tank during critical operations. The minimum volume for the borated water storage tank (390,000 gallons of 1800 ppm boron), as specified in section 3.3, is based on refueling volume requirements and easily satisfies the cold shutdown requirement. The specification assures that the two supplies are available whenever the reactor is critical so that a single failure will not prevent boration to a cold condition. The minimum volumes of boric acid solution given include the boron necessary to account for xenon decay.

The quickest method allows for the necessary boron addition in less than one hour. The primary method of adding boron to the primary system is to pump the concentrated boric acid solution (7,100 ppm boron, minimum) into the makeup tank using the 50 gpm boric acid pumps. Using only one of the two boric acid pumps, the required volume of boric acid can be injected in less than 3.5 hours. The alternate method of addition is to inject boric acid from the borated water storage tank using the high pressure injection pumps.

Concentration of boron in the concentrated boric acid storage tank may be higher than the concentration which would crystallize at ambient conditions. For this reason and to assure a flow of boric acid is available when needed this tank and its associated piping will be kept above 70 F (30 F above the crystallization temperature for the concentration present). Once in the high pressure injection system, the concentrate is sufficiently well mixed and diluted so that normal system temperatures assure boric acid solubility. The value of 70 F is signigicantly above the crystallization temperature for a solution containing 12,200 ppm boron.

REFERENCES

(1) FSAR subsections 9.2 and 9.3

(2) FSAR figure 6.2-1

(3) Technical Specification 3.3

3.3 EMERGENCY CORE COOLING, REACTOR BUILDING EMERGENCY COOLING AND REACTOR BUILDING SPRAY SYSTEMS

Applicability

Applies to the emergency core cooling, Reactor Building emergency cooling and Reactor Building spray systems.

Objective

To define the conditions necessary to assure immediate availability of the emergency core cooling, Reactor Building emergency cooling and Reactor Building spray systems.

Specification

3.3.1 The reactor shall not remain critical, unless the following conditions are met:

A. Injection System

1. The borated water storage tank shall contain a minimum of 390,000 gallons of water having a minimum concentration of 1,800 ppm boron at a temperature not less than 40 F. The manual valves on the discharge line from the borated water storage tank shall be locked open.
2. Two out of three high pressure injection pumps shall be operable.
3. Two safety features actuated decay heat removal pumps shall be operable.
4. Both decay heat removal coolers shall be operable.
5. Two BWST level instrument channels shall be operable.
6. The Reactor Building emergency sump isolation valve shall be either manually or remote-manually operable.
7. One of the two BWST isolation valves shall be open (SFV 25003 or SFV 25004). This valve may be closed during the quarterly valve test specified in the Specifications 4.5.1.2B and 4.5.2.2B.

3.3.1 (Continued)

B. Core Flooding System

1. The two core flooding tanks shall each contain 1040 ± 30 ft^3 of borated water at 600 ± 25 psig.

2. Core flooding tank boron concentration shall not be less than 1,800 ppm boron.

3. The electrically operated discharge valves from the core flood tanks shall be open. The breakers shall be open and so tagged.

4. Two core flood tank pressure instrument channels shall be operable (one per tank minimum).

5. The electrically operated vent valves (HV26511 and HV26512) from the core flood tanks shall be closed. The breakers shall be open and so tagged except during normal venting operations.

C. Reactor Building spray system and Reactor Building emergency cooling system.

The following combination of system components must be operable:

1. Two Reactor Building spray pumps and their associated spray headers with a minimum of 32 percent NaOH solution in the spray additive tanks and,

2. A minimum level of 78 inches of solution shall be available in each spray additive tank.

3. Four emergency cooling units, two with charcoal filter units.

D. Nuclear service cooling and raw water cooling system.

1. Two nuclear service cooling water (NSCW) pumps and two raw water cooling (NSRW) pumps are operable.

2. The manual valves in the NSCW suction and discharge of each operable Reactor Building emergency cooling unit and at the suction of each NSCW pump are locked in their throttled or open position.

3.3.1 (Continued)

D. (Continued)

3. The manual valves in the suction and discharge lines of all operable heat exchangers served by the nuclear service raw water system are locked in their throttled or open position.

E. Safety features valves and interlocks associated with each of the above systems are operable. Inoperable valves shall be placed in the safety position.

3.3.2 Maintenance shall be allowed during power operation on any component(s) in the high pressure, low pressure, nuclear service cooling and raw water cooling, Reactor Building spray, or Reactor Building emergency cooling systems, the core flooding system pressure instrument channels or BWST level channels, which will not degrade safety features system A or B below the level of performance with the single subsystem removed from service. In the context of this specification, a safety features system consists of the following subsystems: high pressure injection, low pressure injection, Reactor Building emergency air cooling, Reactor Building spray, diesel generator, nuclear service cooling water and nuclear service raw water. If the system being repaired is not restored to meet the requirements of specification 3.3.1 within 48 hours, the reactor shall be placed in a hot shutdown condition within 12 hours. If the requirements of specification 3.3.1 are not met within an additional 48 hours, the reactor shall be placed in a cold shutdown condition within 24 hours.

3.3.3 Prior to initiating maintenance on any of the components, the duplicate (redundant) components shall be tested to assure operability, with the component on which maintenance is being performed removed from service.

3.3.4 During power operation, hot standby, hot shutdown or startup conditions, the primary coolant system pressure isolation valves shall be functional as follows:

3.3.4.1 All pressure isolation valves listed in Table 3.3-1 shall be functional as a pressure isolation device, except as specified in 3.3.4.2. Valve leakage shall not exceed the amounts indicated.

3.3.4 (Continued)

3.3.4.2 In the event that integrity of any pressure isolation valve specified in Table 3.3-1 cannot be demonstrated, reactor operation may continue, provided that at least two valves in each high pressure line having a non-functional valve are in and remain in, the mode corresponding to the isolated condition.[a]

3.3.4.3 If Specifications 3.3.4.1 and 3.3.4.2 cannot be met, a shutdown shall be initiated, the reactor shall not remain critical and shall be brought to a cold shutdown condition within an additional 24 hours.

Bases

The requirements of Specification 3.3.1 assure that, before the reactor can be made critical, adequate safety features are operable. Two high pressure injection pumps and two decay heat removal pumps are specified. However, only one of each is necessary to supply emergency coolant to the reactor in the event of a loss-of-coolant accident. Both core flooding tanks are required as a single core flood tank has insufficient inventory to reflood the core.(1)

The borated water storage tank is used for two purposes:

A. As a supply of borated water for accident conditions.

B. As a supply of borated water for flooding the fuel transfer canal during refueling operation.(2)

390,000 gallons of borated water are supplied for emergency core cooling and Reactor Building spray in the event of a loss-of-core coolant accident. This amount fulfills requirements for emergency core cooling. The borated water storage tank minimum volume of 390,000 gallons is based on refueling volume requirements. Heaters maintain the borated water supply at a temperature to prevent freezing. The boron concentration is set at the amount of boron required to maintain the core 1 percent subcritical at 70 F without any control rods in the core. This concentration is 1585 ppm boron while the minimum value specified in the tanks is 1,800 ppm boron.

The requirement that one BWST isolation valve shall be open assures a static head to the injection pump not lined up to the makeup tank.

The post-accident Reactor Building cooling may be accomplished by two spray units or by a combination of two emergency cooling units and one spray unit. The specified requirements assure that the required post accident components are available.

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[a]Motor operated valves shall be placed in the closed position and power supplies deenergized.

3.3 (Continued)

Bases (Continued)

The spray system utilizes common suction lines with the decay heat removal system. If a single train of equipment is removed from either system, the other train must be assured to be operable in each system.
When the reactor is critical, maintenance is allowed per Specification 3.3.2 provided requirements in Specification 3.3.3 are met which assure operability of the duplicate components. Operability of the specified components shall be based on the results of testing as required by Technical Specification 4.5. The maintenance period of up to 48 hours is acceptable if the operability of equipment redundant to that removed from service is demonstrated immediately subsequent to removal. The basis of acceptabil ., is a low likelihood of failure within 48 hours following such demonstration.

In the event that the need for emergency core cooling should occur, functioning of one train (one high pressure injection pump, one decay heat removal pump and both core flooding tanks) will protect the core and in the event of a main coolant loop severance, limit the peak clad temperature to less than 2,200 F and the metal-water reaction to less than 1 percent of the clad.

The nuclear service cooling water system consists of two independent, full capacity, 100 percent redundant systems to ensure continued heat removal.(3)

The requirements of Specification 3.3.4 assure that the decay heat removal system will not be overpressurized, resulting in a LOCA that bypasses containment. Two in-series check valves function as a pressure isolation barrier between the high pressure reactor coolant system and the lower pressure decay heat removal system extending beyond containment. Valve leakage limits provide assurance that the valves are performing their intended isolation function.

REFERENCES

(1) FSAR, paragraph 6.2.1

(2) FSAR, paragraph 9.5.2

(3) FSAR, paragraph 9.4.1

TABLE 3.3-1

PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES

System	Valve No.	Maximum[(a)(b)] Allowable Leakage
Decay Heat Removal	RCS-001 RCS-002 DHS-015 DHS-016	$\leq$ 5.0 GPM for each valve

Footnote:

(a) 1. Leakage rates less than or equal to 1.0 gpm are considered acceptable.

2. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

3. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

4. Leakage rates greater than 5.0 gpm are considered unacceptable.

(b) Minimum test differential pressure shall not be less than 150 psid.

3.4 STEAM AND POWER CONVERSION SYSTEM

Applicability

Applies to the operability of the turbine cycle during normal operation and for the removal of decay heat.

Objective

To specify minimum conditions of the turbine cycle equipment necessary to assure the required steam relief capacity during normal operation and the capability to remove decay heat from the reactor core.

Specification

3.4.1 The reactor shall not remain above 280 F with irradiated fuel in the pressure vessel unless the following conditions are met.

3.4.1.1 Capability to supply feedwater to one steam generator at a process flow rate corresponding to a decay heat of 4-1/2 percent full reactor power from at least one of the following means.

A. A condensate pump and a main feed pump, or

B. A condensate pump or

C. An auxiliary feedwater pump.

The required flow rates are:

Feedwater temperature, degrees F	Required flow, gpm
40	743
60	756
90	780

3.4.1.2 Two steam system safety valves are operable per steam generator.

3.4.1.3 The turbine bypass system to the condenser shall have one valve operable or the atmospheric dump system shall have a minimum of 1 of 3 valves operable per steam generator.

3.4.1.4 A minimum of 250,000 gallons of water shall be available in the condensate storage tank.

3.4.2 In addition to the requirements of 3.4.1, the reactor shall not remain critical unless the following condtions are met:

3.4.2.1 Seventeen of the eighteen main steam system safety valves are operable.

3.4.2.2 When two independent 100% capacity auxiliary feedwater flow paths are not available, the capacity shall be restored within 72 hours or the plant shall be placed in a cooling mode which does not rely on steam generators for cooling within the next 12 hours.

3.4.2.3 When at least one 100% capacity auxiliary feedwater flow path is not available, the reactor shall be made subcritical within four hours and the facility placed in a shutdown cooling mode which does not rely on steam generators for cooling within next 12 hours.

Bases

The feedwater system and the turbine bypass system are normally used for decay heat removal and cooldown above 280 F. Feedwater makeup is supplied by operation of a condensate pump and main feedwater pump. In the event of complete loss of electrical power, feedwater is supplied by a turbine driven auxiliary feedwater pump which takes suction from the condensate storage tank. Steam relief would be through the system's atmospheric relief valves.

If neither main feed pump is available, feedwater can be supplied to the steam generators by an auxiliary feedwater pump and steam relief would be through the turbine bypass system to the condenser.

In order to heat the reactor coolant system above 280 F the maximum steam removal capability required is 4-1/2 percent of rated power. This is the maximum decay heat rate at 30 seconds after a reactor trip. The requirement for two steam system safety valves per steam generator provides a steam relief capability of over 10 percent per steam generator (1,341,938 lb/h). In addition, two turbine bypass valves to the condenser or two atmospheric dump valves will provide the necessary capacity.

The 250,000 gallons of water in the condensate storage tank is the amount needed for cooling water to the steam generators for a period in excess of one day following a complete loss of all unit ac power.(1)

The minimum relief capacity of seventeen steam system safety valves is 13,329,163 lb/hr.(2) This is sufficient capacity to protect the steam system under the design overpower condition of 112 percent.(3)

REFERENCES

(1) FSAR paragraph 14.1.2.8.4

(2) FSAR paragraph 10.3.4

(3) FSAR Appendix 3A, Answer to Question 3A.5

3.5 INSTRUMENTATION SYSTEMS

3.5.1 Operational Safety Instrumentation

Applicability

Applies to unit instrumentation and control systems.

Objective

To delineate the conditions of the unit instrumentation and safety circuits necessary to assure reactor safety.

Specifications

3.5.1.1 Startup and operation are not permitted unless the requirements of table 3.5.1-1, Columns A and B are met.

3.5.1.2 In the event the number of protection channels operable falls below the limit given under table 3.5.1-1, Columns A and B, operation shall be limited as specified in Column C.

In the event the total number of operable Process Instrumentation channels is less than the Total Number of Channel(s), restore the inoperable channels to operable status within 7 days, or be in at least hot shutdown within the next 12 hours. If the number of operable channels is less than the minimum channels operable, either restore the inoperable channels to operable within 48 hours or be in at least hot shutdown within the next 12 hours.

3.5.1.3 For on-line testing or in the event of a protection instrument or channel failure, a key operated channel bypass switch associated with each reactor protection channel will be used to lock the channel trip relay in the untripped state as indicated by a light. Only one channel shall be locked in this untripped state at any one time.

3.5.1.4 The key operated shutdown bypass switch associated with each reactor protection channel shall not be used during reactor power operation.

3.5.1.5 During startup when the intermediate range instrument comes on scale, the overlap between the intermediate range and the source range instrumentation shall not be less than one decade. If the overlap is less than one decade, the flux level shall be maintained in the source range until the one decade overlap is achieved.

Specifications (Continued)

3.5.1.6 In the event that one of the trip devices in either of the sources supplying power to the control rod drive mechanisms fails in the untripped state, the power supplied to the rod drive mechanisms through the failed trip device shall be manually removed within 30 minutes. The condition will be corrected and the remaining trip devices shall be tested within eight hours. If the condition is not corrected and the remaining trip devices are not tested within the eight-hour period, the reactor shall be placed in the hot shutdown condition within an additional four hours.

Bases

Every reasonable effort will be made to maintain all safety instrumentation in operation. A startup is not permitted unless three power range neutron instrument channels and two channels each of the following are operable: four reactor coolant temperature instrument channels, four reactor coolant flow instrument channels, four reactor coolant pressure instrument channels, four pressure-temperature instrument channels, four flux-imbalance flow instrument channels, four power-number of pumps instrument channels, and four high reactor building pressure instrument channels. The safety features actuation system must have two analog channels functioning correctly prior to startup.

Operation at rated power is permitted as long as the systems have at least the redundancy requirements of Column B (Table 3.5.1-1). This is in agreement with redundancy and single failure criteria of IEEE 279 as described in FSAR section 7.

There are four reactor protection channels. Normal trip logic is two out of four. Required trip logic for the power range instrumentation channels is two out of three. Minimum trip logic on other instrumentation channels is one out of two.

The four reactor protection channels were provided with key operated bypass switches interlocked to allow on-line testing or maintenance on only one channel at a time during power operation. Each channel is provided alarm and lights to indicate when that channel is bypassed.

Each reactor protection channel key operated shutdown bypass switch is provided with alarm and lights to indicate when the shutdown bypass switch is being used. There are four shutdown bypass keys in the control room under the administrative control of the shift supervisor. The keys will not be used during reactor power operation.

The source range and intermediate range nuclear flux instrumentation scales overlap by one decade. This decade overlap will be achieved at 10^{-10} amps on the intermediate range scale.

3.5.1 (Continued)

Bases (Continued)

Power is normally supplied to the control rod drive mechanisms from two separate parallel 480 volt sources. Redundant trip devices are employed in each of these sources. If any one of the trip devices fails in the untripped state on-line repairs to the failed device, when practical, will be made, and the remaining trip devices will be tested. Eight hours is ample time to test the remaining trip devices and in many cases make on-line repairs.

The OPERABILITY of the SFAS instrumentation systems and bypasses ensure that:

1) the associated SFAS action will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint,

2) the specified coincidence logic is maintained,

3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and

4) sufficient system functional capability is available for SFAS purposes from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses.

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident", December 1975 and NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."

REFERENCE

FSAR, Subsection 7.1

Table 3.5.1-1

INSTRUMENTS OPERATING CONDITIONS

Functional Unit	(A) Minimum Operable Channels	(B) Minimum Degree of Redundancy	(C) Operator Action if Conditions of Columns A and B Cannot be Met
Reactor Protection System			
.1 Manual pushbutton	1	0	Bring to hot shutdown within 12 hours
.2 Power range instrument channel	3(a)	1(a)	Bring to hot shutdown within 12 hours
.3 Intermediate range instrument channels	1	0	Bring to hot shutdown within 12 hours (b)
.4 Source range instrument channels	1	0	Bring to hot shutdown within 12 hours (c)
.5 Reactor coolant temperature instrument channels	2	1	Bring to hot shutdown within 12 hours
.6 Pressure-temperature instrument channels	2	1	Bring to hot shutdown within 12 hours
.7 Flux/imbalance/flow instrument channels	2	1	Bring to hot shutdown within 12 hours

(a) For channel testing, calibration or maintenance, the minimum number of operable channels may be two and a degree of redundancy of one for a maximum of 4 hours.

(b) When 2 of 4 power range instrument channels are greater than 10 percent full power, hot shutdown is not required.

(c) When 1 of 2 intermediate range instrument channels is greater that 10^{-10} amps, or 2 of 4 power range instrument channels are greater than 10 percent full power, hot shutdown is not required.

Table 3.5.1-1 (Continued)

INSTRUMENTS OPERATING CONDITIONS

Functional Unit	(A) Minimum Operable Channels	(B) Minimum Degree of Redundancy	(C) Operator Action if Conditions of Columns A and B Cannot be Met
Reactor Protection System			
.8 Reactor coolant pressure			
(a) High reactor coolant pressure instrument channels	2	1	Bring to hot shutdown within 12 hours
(b) Low reactor coolant pressure instrument channels	2	1	Bring to hot shutdown within 12 hours
.9 Power/number of pumps instrument channels	2	1	Bring to hot shutdown within 12 hours
.10 High Reactor Buidling pressure channels	2	1	Bring to hot shutdown within 12 hours
.11 Loss of Main Feedwater	2	1	Bring to hot shutdown within 12 hours
.12 Turbine/Generator	1	0	Bring to hot shutdown within 12 hours
Safety Features*			
.1 High pressure injection, Reactor Building Isolation, and Reactor Building emergency cooling			
a. Reactor coolant pressure instrument channels	2	1	Bring to hot shutdown within 12 hours

* If minimum conditions are not met within 48 hours after hot shutdown, the unit shall be placed in a cold shutdown condition within an additional 24 hours.

Table 3.5.1-1 (Continued)

INSTRUMENTS OPERATING CONDITIONS

Functional Unit	(A) Minimum Operable Channels	(B) Minimum Degree of Redundancy	(C) Operator Action if Conditions of Columns A and B Cannot be Met
<u>Safety Features</u>*			
b. Reactor Building Pressure instrument channels	2	1	Bring to hot shutdown within 12 hours
c. Manual pushbutton	2	1	Bring to hot shutdown within 12 hours
d. Automatic Actuation Logic	2	1	Bring to hot shutdown within 12 hours
2. Low pressur injection			
a. Reactor coolant pressure instrument channels	2	1	Bring to hot shutdown within 12 hours
b. Reactor Building pressure instrument channels	2	1	Bring to hot shutdown within 12 hours
c. Manual pushbutton	2	1	Bring to hot shutdown within 12 hours
3. Reactor Building spray pump			
a. Reactor Building instrument channel	2	1	Bring to hot shutdown within 12 hours
b. Manual pushbutton	2	1	Bring to hot shutdown within 12 hours
4. Reactor Building spray valve			
a. Reactor Building pressure instrument channels	2	1	Bring to hot shutdown within 24 hours
b. Manual pushbutton	2	1	Bring to hot shutdown within 24 hours

* If minimum conditions are not met within 48 hours after hot shutdown, the unit shall be placed in a cold shutdown condition within an additional 24 hours.

Table 3.5.1-1 (Continued)

INSTRUMENTS OPERATING CONDITIONS

Functional Unit	(A) Total Number of Channels	(B) Minimum Channels Operable	(C) Operator Action if Conditions of Columns A and B Cannot be Met
Process Instrumentation			
1. Pressurizer Water Level	3	1	See Section 3.5.1.2
2. Auxiliary Feedwater Flow**	1 per loop	1 per loop	
3. Reactor Coolant System Sub-cooling Margin Monitor	2	1	
4. EMOV Position Indicator (Primary Detector) power indicator***	1/valve	1/valve	
5. EMOV Position Indicator (Backup Detector) acoustic or T/C***	1/valve	0	
6. EMOV Block Valve Position Indicator***	1/valve	1/valve	
7. Safety Valve Position Indicator (Primary Detector) T/C	1/valve	1/valve	
8. Safety Valve Position Indicator (Backup Detector) acoustic	1/valve	0	
Auxiliary Feedwater			
1. Low Main Feedwater Pressure: Start Motor Driven Pump and Turbine Drive Pump	2	1	
2. Contact Monitor - RCS Pumps: Start Motor Driven and Turbine Driven Pumps	2	1	

** OTSG level may be used for flow.

*** Applies when EMOV is OPERABLE.

3.5.2 Control Rod Group and Power Distribution Limits

Applicability

This specification applies to power distribution and operation of control rods during power operation.

Objective

To assure an acceptable core power distribution during power operation, to set a limit on potential reactivity insertion from a hypothetical control rod ejection, and to assure core subcriticality after a reactor trip.

Specification

3.5.2.1 The available shutdown margin shall be not less than 1 percent Δ k/k with the highest worth control rod fully withdrawn.

3.5.2.2 Operation with inoperable rods:

A. Operation with more than one inoperable rod as defined in Specification 4.7.1 and 4.7.2.3 in the safety or regulating rod banks shall not be permitted.

B. If a control rod in the regulating and/or safety rod banks is declared inoperable in the withdrawn position as defined in Specification paragraph 4.7.1.1 and 4.7.1.3, an evaluation shall be initiated immediately to verify the existence of 1 percent Δ k/k hot shutdown margin. Boration may be initiated to increase the available rod worth either to compensate for the worth of the inoperable rod or until the regulating banks are fully withdrawn, whichever occurs first.

C. If within one hour of determination of an inoperable rod as defined in Specification 4.7.1, it is not determined that a 1 percent Δ k/k hot shutdown margin exists combining the worth of the inoperable rod with each of the other rods, the reactor shall be brought to the hot standby condition until this margin is established.

D. Following the determination of an inoperable rod as defined in Specification 4.7.1, all rods shall be exercised by a movement until indication is noted but not exceeding 2 inches within 24 hours and exercised weekly until the rod problem is solved.

3.5.2 (Continued)

E. If a control rod in the regulating or safety rod groups is declared inoperable per 4.7.1.2, power shall be reduced to 60 percent of the thermal power allowable for the reactor coolant pump combination.

F. If a control rod in the regulating or axial power shaping groups is declared inoperable per Specification 4.7.1.2, operation above 60 percent of rated power may continue provided the rods in the group are positioned such that the rod that was declared inoperable is maintained within allowable group average position limits of Specification 4.7.1.2 and the withdrawal limits of Specification 3.5.2.5.c.

3.5.2.3 The worth of a single inserted control rod shall not exceed 0.65 percent $\Delta k/k$ at rated power or 1.0 percent $\Delta k/k$ at hot zero power except for physics testing when the requirement of Specification 3.1.8 shall apply.

3.5.2.4 Quadrant Power Tilt

A. With the Quadrant Power Tilt determined to exceed 4.92 percent but less than or equal to 11.07 percent except for physics test.

1. Within 2 hours:

a) Either reduce the quadrant power tilt to $\leq$ 4.92 percent, or

b) Reduce thermal power so as not to exceed thermal power, including power level cutoff, allowable for the reactor coolant pump combination, less at least 2 percent for each 1 percent, or fraction thereof, of quadrant power tilt in excess of 4.92 percent. Within 4 hours, take action to reduce the high flux trip and flux-Δflux-flow trip setpoints at least 2 percent for each 1 percent, or fraction thereof, of quadrant power tilt in excess of 4.92 percent.

2. Verify that the Quadrant Power Tilt is $\leq$ 4.92 percent within 24 hours after exceeding that limit or reduce Thermal Power to less than 60 percent of Thermal Power allowable for the reactor coolant pump combination within the next 2 hours and reduce the High Flux Trip Setpoint to $\leq$ 65.5 percent of Thermal Power allowable for the reactor coolant pump combination within the next 4 hours.

3.5.2 (Continued)

3. Identify and correct the cause of the out of limit condition prior to increasing Thermal Power; subsequent Power Operation above 60 percent of Thermal Power allowable for the reactor coolant pump combination may proceed provided that the Quadrant Power Tilt is verified $\leq$ 4.92 percent at least once per hour for 12 hours or until verified acceptable at 95 percent or greater Rated Thermal Power.

B. With the Quadrant Power Tilt determined to exceed 11.07 percent but less than 20 percent, due to misalignment of either a safety, regulating or axial power shaping rod:

1. Reduce thermal power at least 2 percent for each 1 percent, or fraction thereof, of Quadrant Power Tilt in excess of 4.92 percent within 30 minutes.

2. Verify that the Quadrant Power Tilt is $\leq$ 11.07 percent within 2 hours after exceeding that limit or reduce Thermal Power to less than 60 percent of Thermal Power allowable for the Reactor Coolant Pump combination within the next 2 hours and reduce the High Flux Trip Setpoint to $\leq$ 65.5 percent of Thermal Power allowable for the Reactor Coolant Pump combination within the next 4 hours.

3. Identify and correct the cause of the out of limit condition prior to increasing Thermal Power; subsequent Power Operation above 60 percent of Thermal Power allowable for the Reactor Coolant Pump combination may proceed provided that the Quadrant Power Tilt is verified $\leq$ 4.92 percent at least once per hour for 12 hours or until verified acceptable at 95 percent or greater Rated Thermal Power.

C. With the Quadrant Power Tilt determined to exceed 11.07 percent but less than 20 percent, due to causes other than the misalignment of either a safety, regulating or axial power shaping rod:

1. Reduce Thermal Power to less than 60 percent of Thermal Power allowable for the Reactor Coolant Pump combination within 2 hours and reduce the High Flux Trip Setpoint to $\leq$ 65.5 percent of Thermal Power allowable for the Reactor Coolant Pump combination within the next 4 hours.

3.5.2 (Continued)

2. Identify and correct the cause of the out of limit condition prior to increasing Thermal Power; subsequent Power Operation above 60 percent of Thermal Power allowable for the Reactor Coolant Pump combination may proceed provided that the Quadrant Power Tilt is verified $\leq$ 4.92 percent at least once per hour for 12 hours or until verified at 95 percent or greater Rated Thermal Power.

D. With the Quadrant Power Tilt determined to exceed 20 percent, a controlled shutdown shall be initiated immediately and the reactor shall be brought to the hot shutdown condition within 4 hours.

E. Whenever the reactor is brought to hot shutdown pursuant to 3.5.2.4.D above, subsequent reactor operation is permitted for the purpose of measurement, testing and corrective action provided the thermal power and power range high flux setpoint allowable for the Reactor Coolant Pump combination are restricted by a reduction of 2 percent of maximum allowable power for each 1 percent tilt, or fraction thereof, for the maximum tilt observed prior to shutdown.

F. The Quadrant Power Tilt shall be determined to be within the limits at least once every shift during operation above 15 percent of Rated Thermal Power except when the Quadrant Power Tilt alarm is inoperable, then the Quadrant Power Tilt shall be calculated and evaluated at least once every 2 hours.

3.5.2.5 Control Rod Positions

A. Technical Specifiction 3.1.3.5 (safety rod withdrawal) does not prohibit the exercising of individual safety rods as required by Table 4.1-2 or apply to inoperable safety rod limits in Technical Specification 3.5.2.2.

B. Operating rod group overlap shall be 25 percent, $\pm$5 percent between three sequential groups, except for physics tests.

C. Position limits are specified for regulating and axial power shaping control rods. Except for physics tests or exercising control rods, the regulating control rod insertion/withdrawal limits are specified on Figures 3.5.2-1 through 3.5.2-6. Also excepting physics tests or exercising control rods, the axial

3.5.2 (Continued)

power shaping control rod insertion/withdrawal limits are specified on Figures 3.5.2-7 and 3.5.2-8. If any of these control rod position limits are exceeded, corrective measures shall be taken immediately to achieve an acceptable control rod position. Acceptable control rod positions shall be attained within two hours.

D. Except for physics test, power shall not be increased above the power level cutoff of 92 percent of the maximum allowable power level unless one of the following conditions is satisfied:

1. Xenon reactivity is within 10 percent of the equilibrium value for operation at the maximum allowable power level and asymptotically approaching stability.

2. Except for Xenon free startup, when 3.5.2.5D(1) applies, the reactor has operated within a range of 87 to 92 percent of the maximum allowable power for a period exceeding 2 hours in the soluble poison control mode.

3.5.2.6 Reactor power imbalance shall be monitored on a frequency not to exceed two hours during power operation above 40 percent rated power. Except for physics test, imbalance shall be maintained within the envelope defined by Figures 3.5.2-9, 3.5.2-10 and 3.5.2-11. If the imbalance is not within the envelope defined by Figures 3.5.2-9, 3.5.2-10 and 3.5.2-11, corrective measures shall be taken to achieve an acceptable imbalance. If an acceptable imbalance is not achieved within two hours, reactor power shall be reduced until imbalance limits are met.

3.5.2.7 The control rod drive patch panels shall be locked at all times with limited access to be authorized by the superintendent or his designated representative.

Bases

The power-imbalance envelope defined in Figures 3.5.2-9, 3.5.2-10 and 3.5.2-11 are based on LOCA analyses which have defined the maximum linear heat rate such that the maximum clad temperature will not exceed the Final Acceptance Criteria.(3) Corrective measures will be taken should the indicated quadrant tilt, rod position, or inbalance be outside their specified boundry. Operation in a situation that would cause the Final Acceptance Criteria to be approached should a LOC occur is highly improbable because all of the power distribution parameters (quadrant tilt, rod

3.5.2 (Continued)

Bases (Continued)

position, and imbalance) must be at their limits while simultaneously all other engineering and uncertainty factors are also at their limits.**

A. Nuclear uncertainty factors.
B. Thermal calibration uncertainty.
C. Hot rod manufacturing tolerance factors
D. Fuel densification effects.

The conservative application of the above peaking augmentation factors compensates for the potential peaking penalty due to Fuel rod bow.

The 25+ percent overlap between successive control rod groups is allowed since the worth of a rod is lower at the upper and lower part of the stroke. Control rods are arranged in groups defined as follows:

Group	Function
1	Safety
2	Safety
3	Safety
4	Safety
5	Regulating
6	Regulating
7	Regulating
8	APSR (axial power shaping bank)

The rod position limits are based on the most limiting of the following three criteria: ECCS power peaking, shutdown margin, and potential ejected rod worth. Therefore, compliance with the ECCS power peaking criterion is ensured by the rod position limits. The minimum available rod worth, consistent with the rod position limits, provides for achieving hot shutdown by reactor trip at any time, assuming the highest worth control rod that is withdrawn remains in the full out position.(1) The rod position limits

** Actual operating limits depend on whether or not incore or excore detectors are used and their respectivce instrument calibration errors. The method used to define the operating limits is defined in plant operating procedures.

3.5.2 (Continued)

Bases (Continued)

also ensure that inserted rod groups will not contain single rod worths greater than 0.65 percent $\Delta k/k$ at rated power. These values have been shown to be safe by the safety analysis of hypothetical rod ejection accident.(2) A maximum single inserted control rod worth of 1.0 percent $\Delta k/k$ is allowed by the rod position limits at hot zero power. A single inserted control rod worth of 1.0 percent $\Delta k/k$ at beginning-of-life, hot zero power would result in a lower transient peak thermal power and, therefore, less severe environmental consequences than an 0.65 percent $\Delta k/k$ ejected rod worth at rated power.

Control rod groups are withdrawn in sequence beginning with Group 1. Groups 5, 6, and 7 are overlapped 25 percent. The normal position at power is for Group 7 to be partially inserted.

The Quadrant Power Tilt limits set forth in Specification 3.5.2.4 have been established to prevent the linear heat rate peaking increase associated with a positive quadrant power tilt during normal power operation from exceeding 7.36 percent. The limits in Specification 3.5.2.4 are measurement system independent. The actual operating limits, with the appropriate allowance for observability and instrumentation errors, for each measurement system are defined in the station operation procedures.

The Quadrant Tilt and axial imbalance monitoring in Specifications 3.5.2.4F and 3.5.2.6, respectively, normally will be performed in the process computer. The two-hour frequency for monitoring these qualities will provide adequate surveillance when the computer is out of service.

Allowance is provided for withdrawal limits and reactor power imbalance limits to be exceeded for a period of two hours without specification violation. Acceptable rod positions and imbalance must be achieved within the two-hour time period or appropriate action such as a reduction of power taken.

Operating restrictions are included in Technical Specifications 3.5.2.5.D 1 and 3.5.2.5.D.2 to prevent excessive power peaking by transient xenon. The xenon reactivity must either be beyond the "undershoot" region and asymptotically approaching its equilibrium value at rated power or the reactor must be operated in the range of 87% to 92% of the maximum allowable power for a period exceeding two hours in the soluble poison control mode so that the transient peak is burned out at a lower power level.

REFERENCES

(1) FSAR, Section 3.2.2.1.2
(2) FSAR, Section 14.2.2.4
(3) BAW-1499 September 1978, Page 7-3

FIGURE 3.5.2-1 Rod Index Vs Power Level
Rancho Seco Cycle 5
0 - 60 EFPD
Four Pump Operation

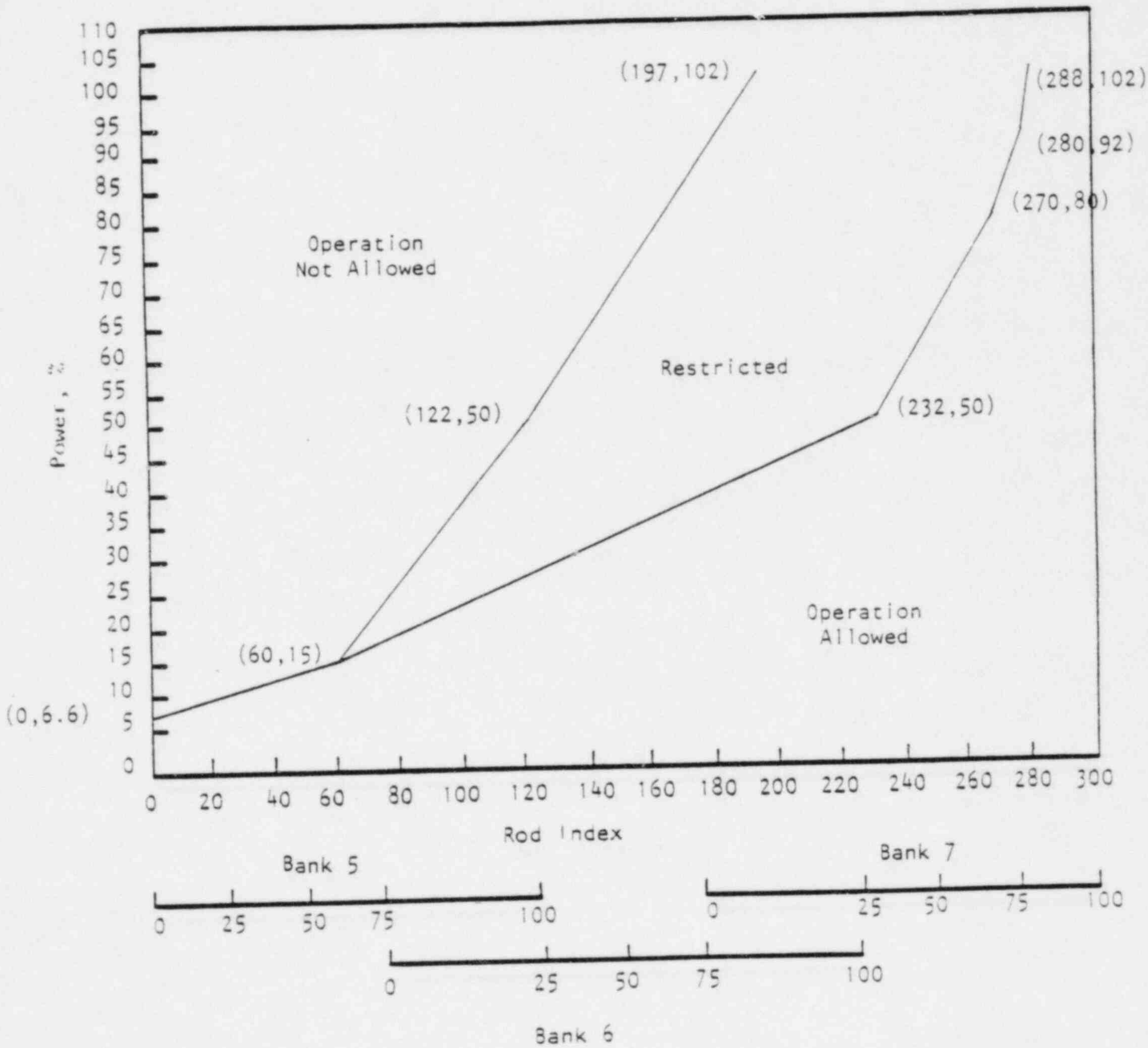


FIGURE 3.5.2-2 Rod Index Vs Power Level
Rancho Seco Cycle 5
50 - 250 EFPD
Four Pump Operation

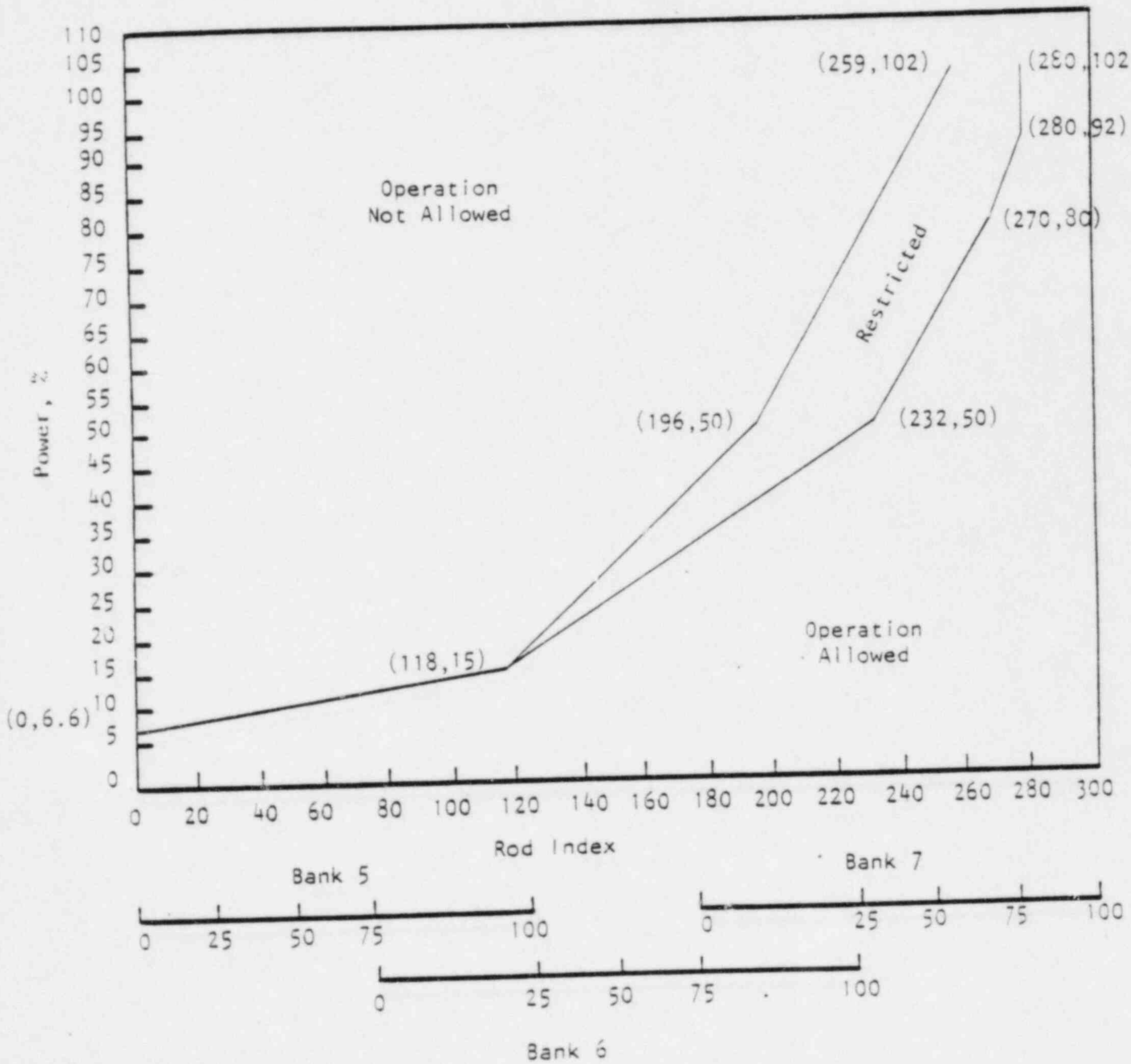


FIGURE 3.5.2-3 Rod Index Vs Power Level
Rancho Seco Cycle 5
230 EFPD - EOC (APSR Out)
Four Pump Operation

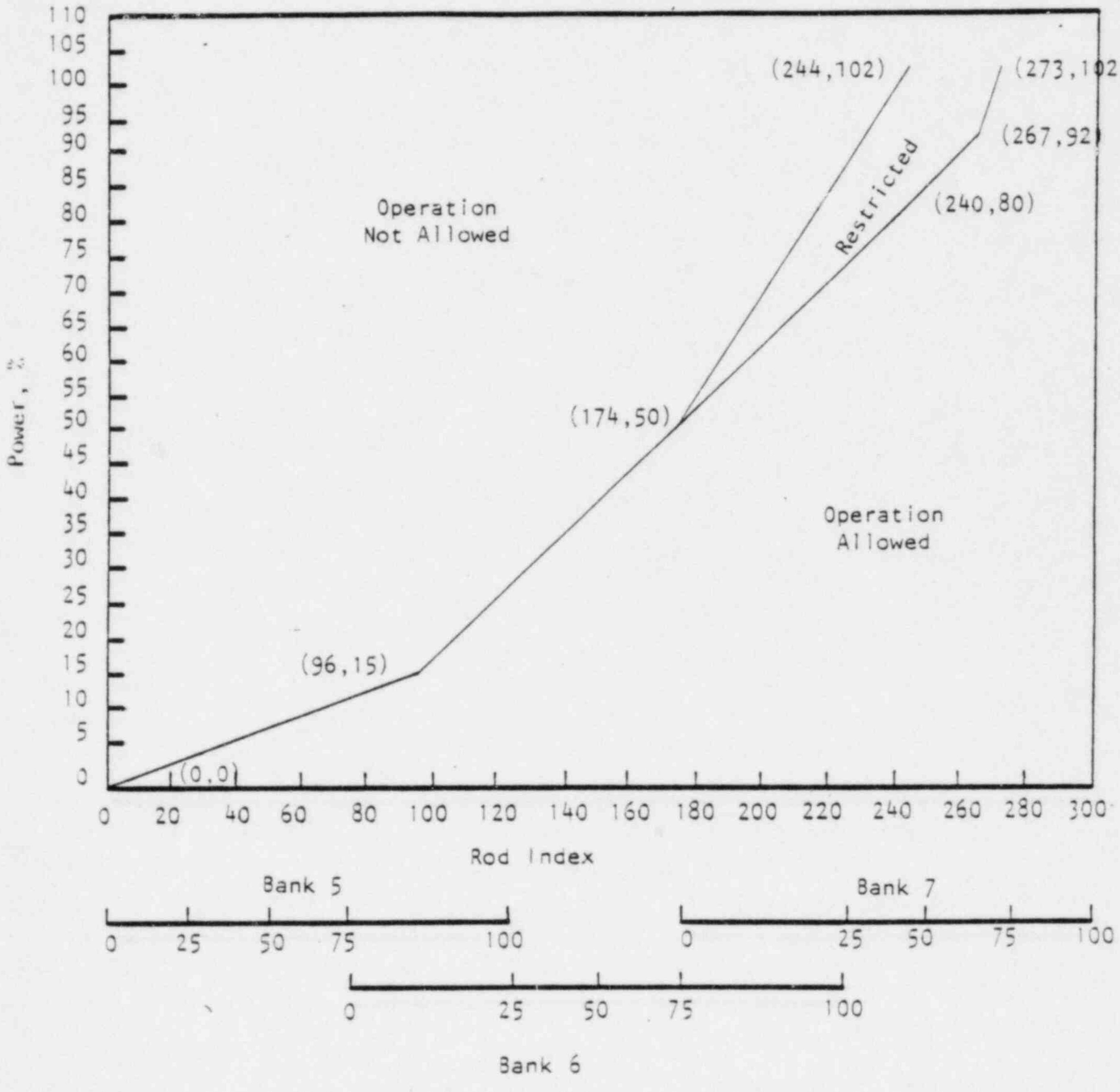


FIGURE 3.5.2-4 Rod Index Vs Power Level
Rancho Seco Cycle 5
0 - 60 EFPD
3 Pump Operation

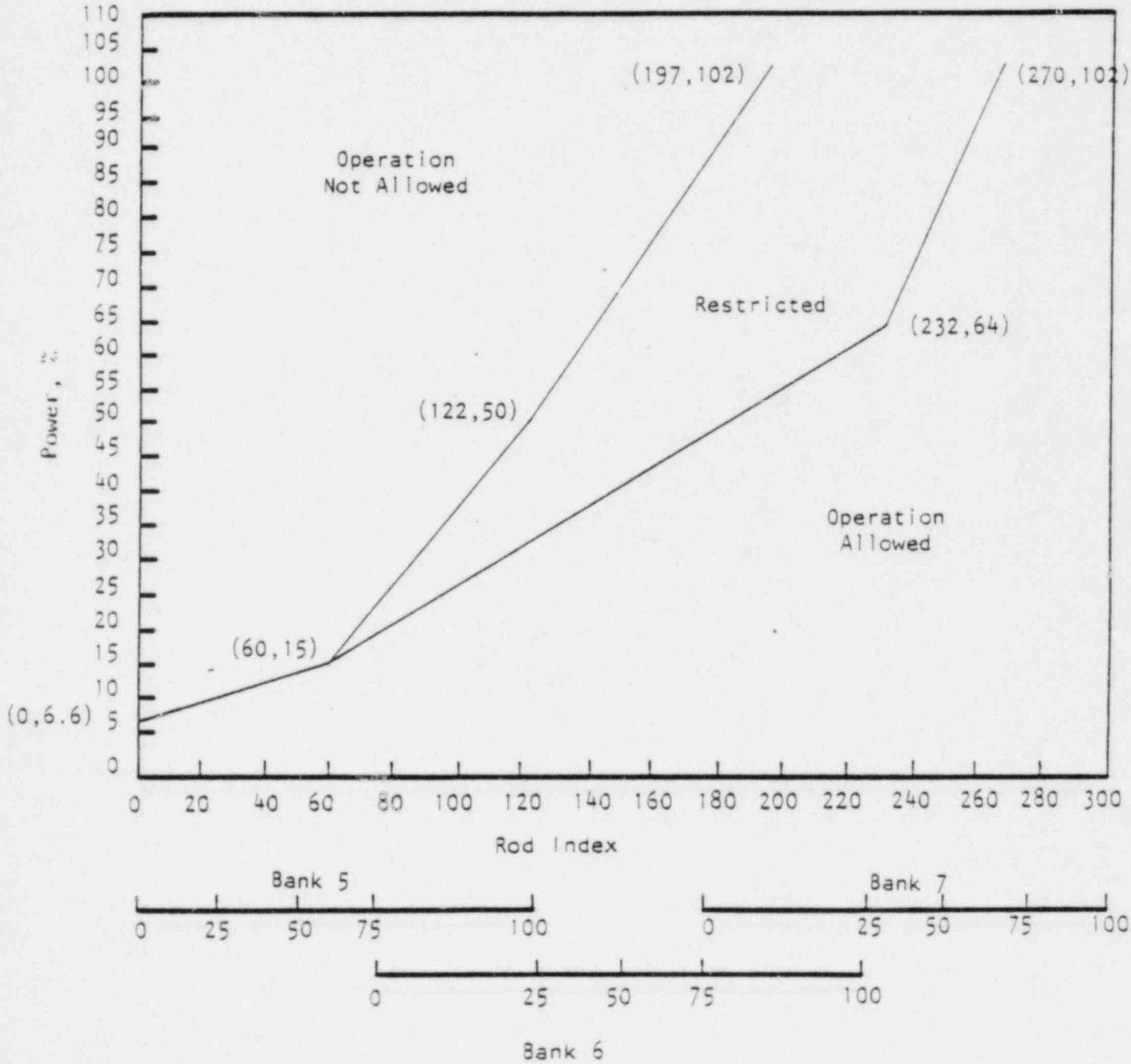


FIGURE 3.5.2-5 Rod Index Vs Power Level
Rancho Seco Cycle 5
50 - 250 EFPD
3 Pump Operation

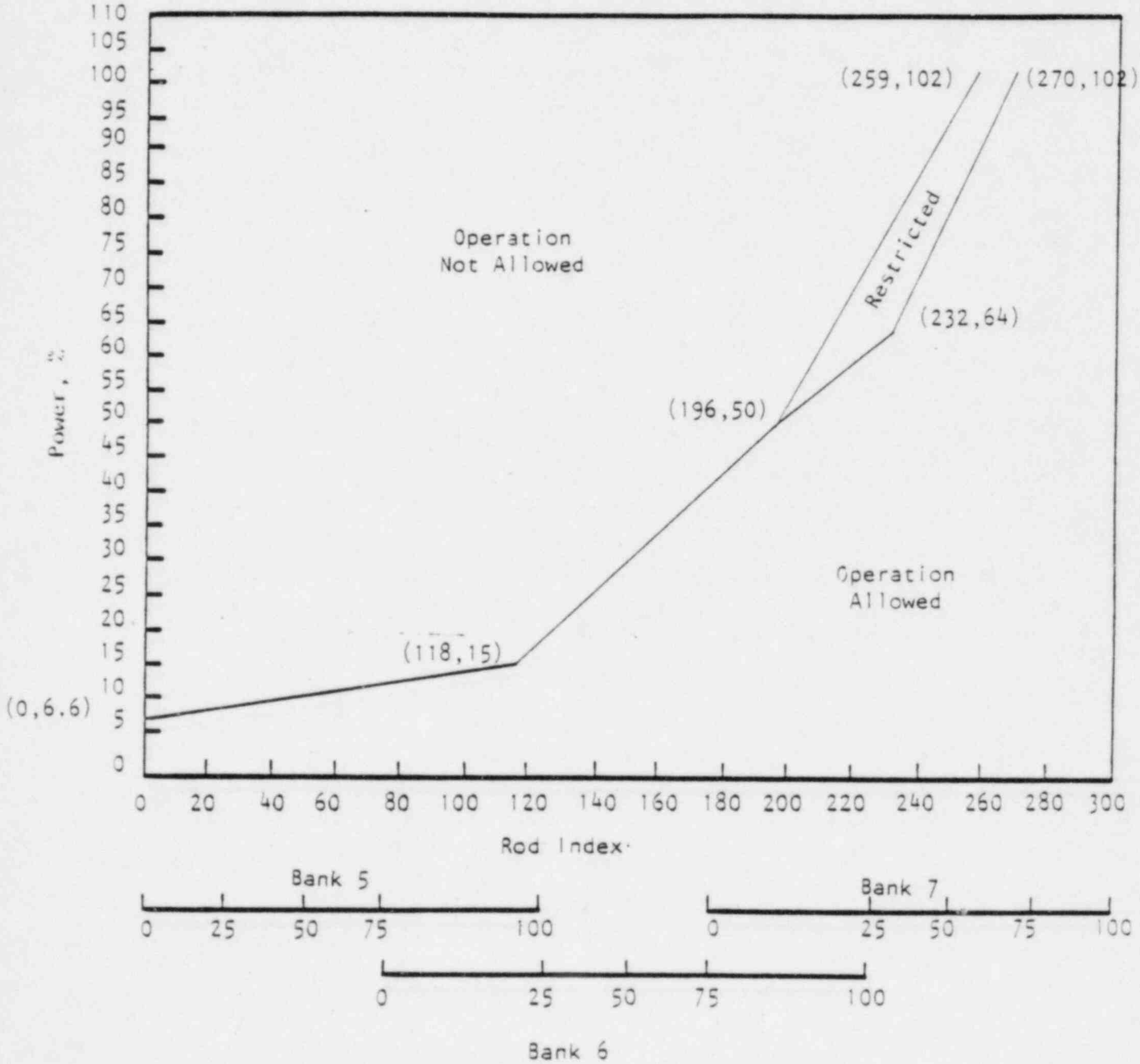


FIGURE 3.5.2-6 Rod Index Vs Power Level
Rancho Seco Cycle 5
230 EFPD - EOC (APSR Out)
3 Pump Operation

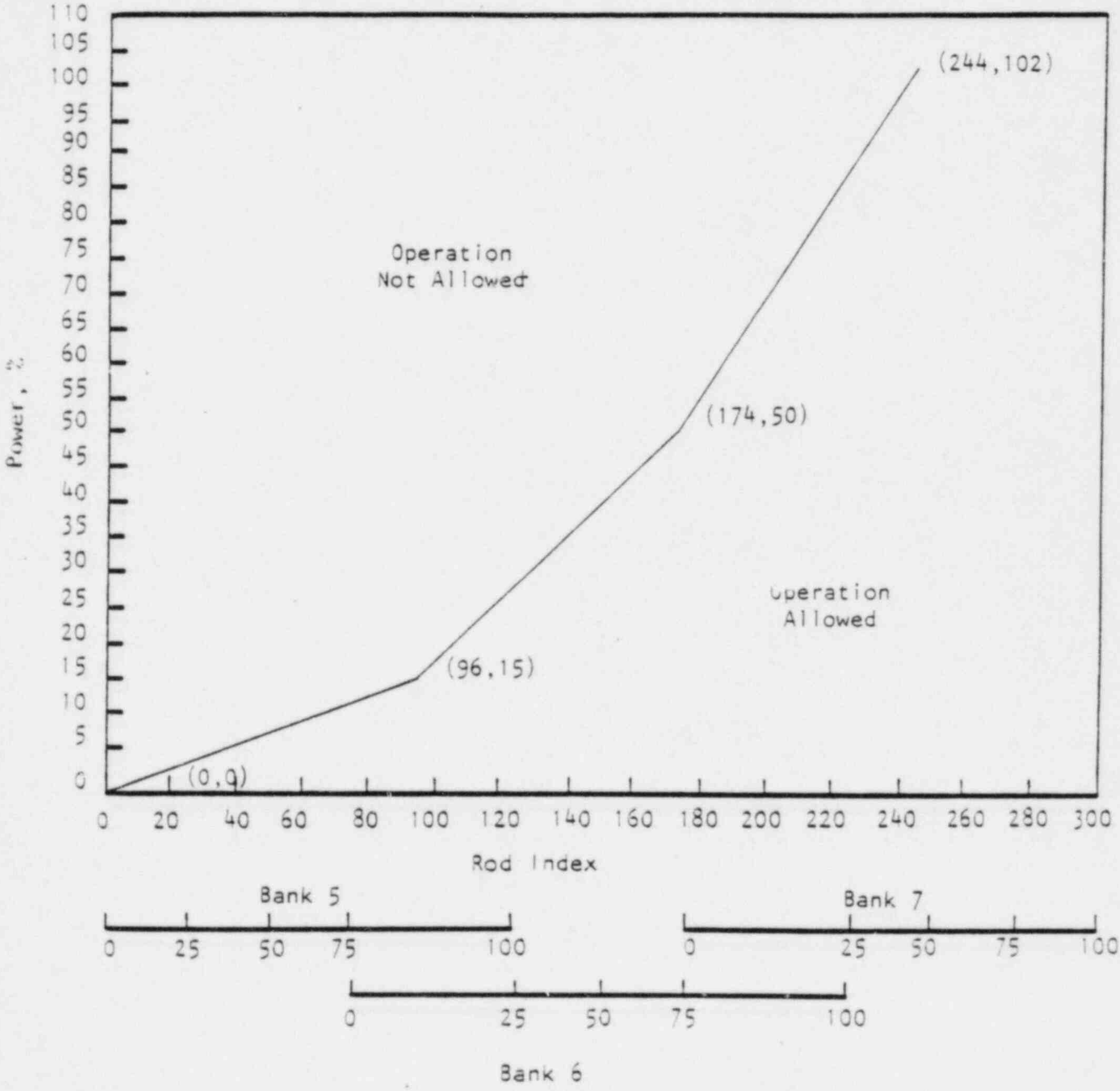


FIGURE 3.5.2-7 APSR Withdrawal Vs Power Level
Rancho Seco Cycle 5
APSR Position Limits (0-60 EFPD)

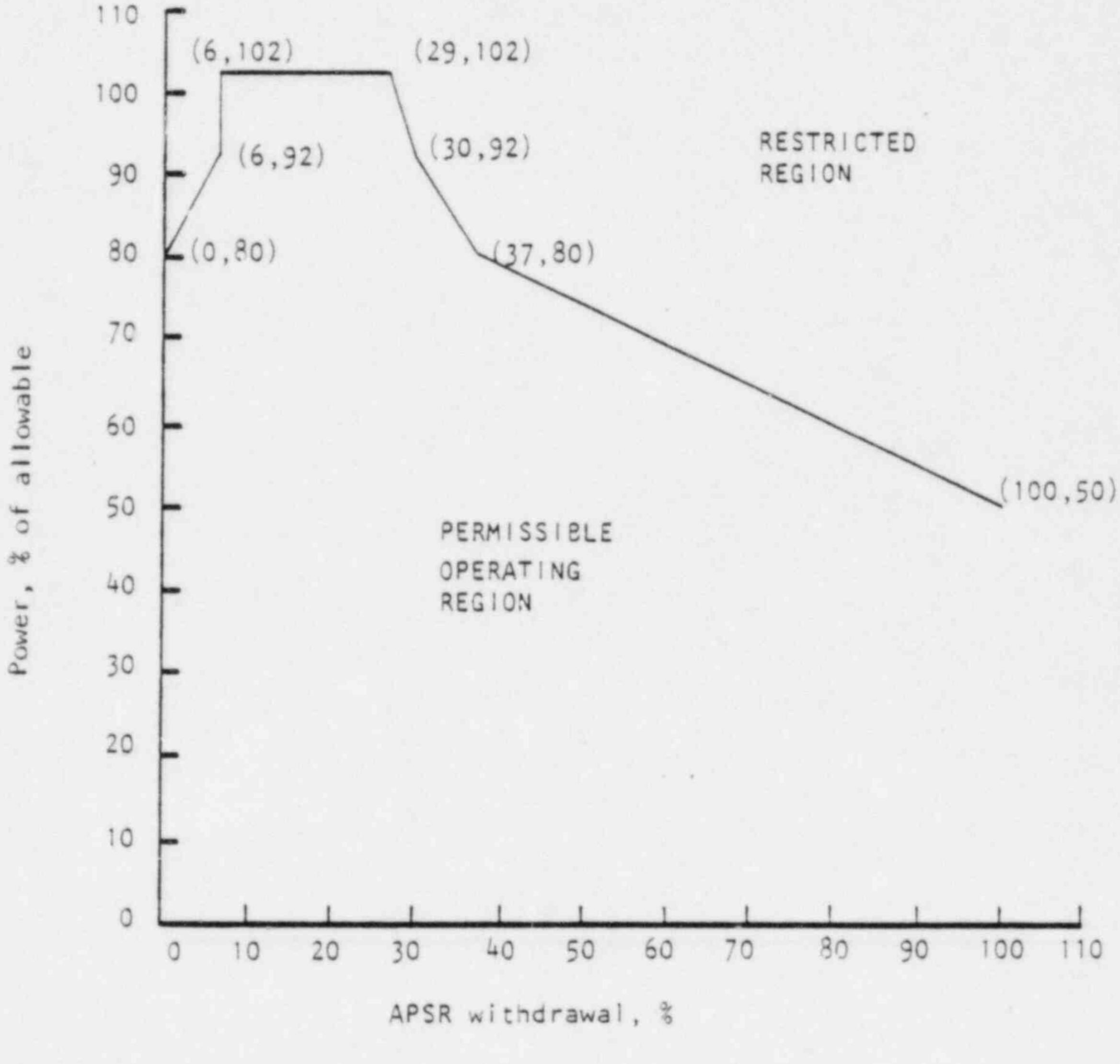


FIGURE 3.5.2-8 APSR Withdrawal Vs Power Level
Rancho Seco Cycle 5
APSR Position Limits (50-250 EFPD)

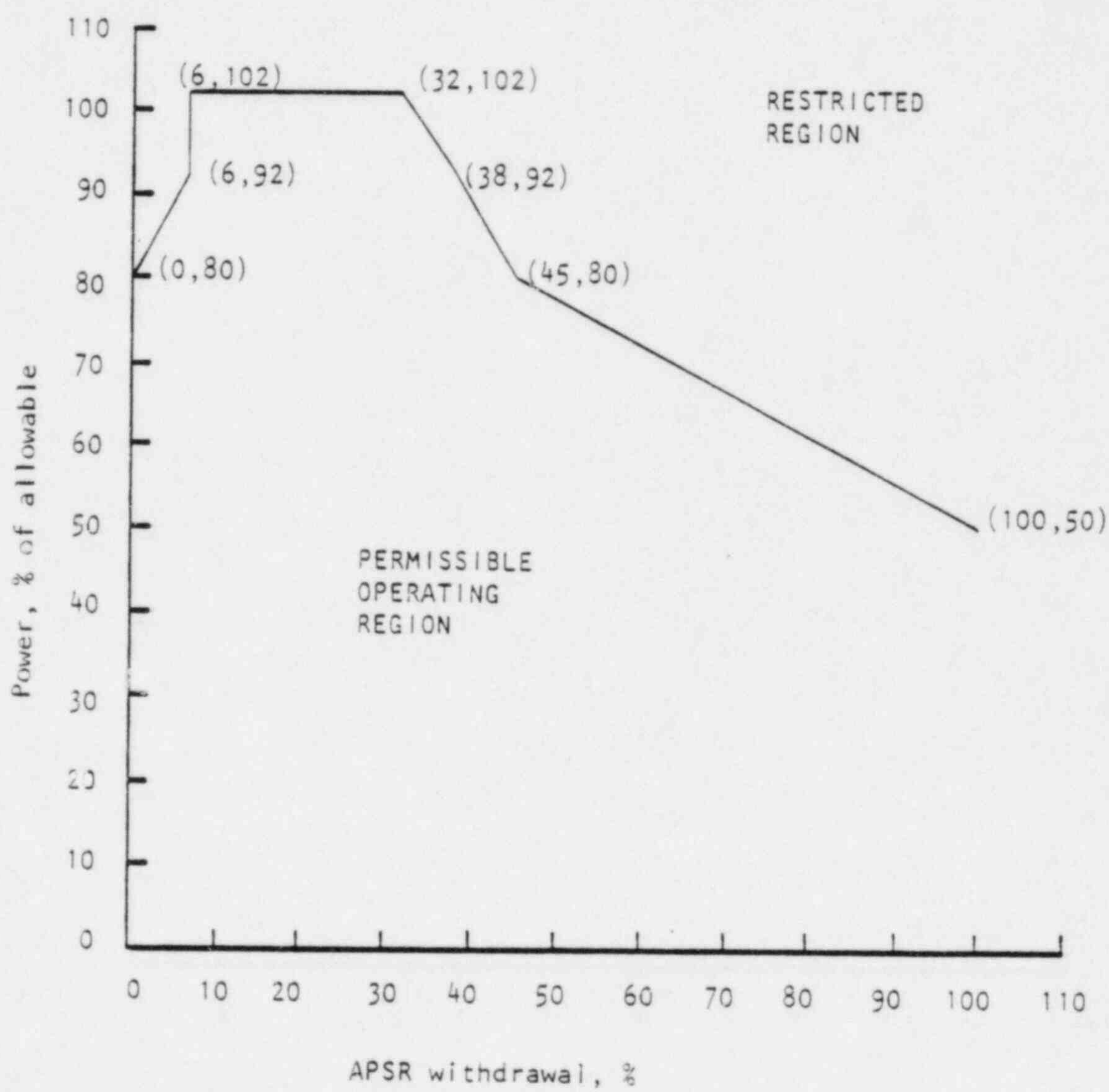


FIGURE 3.5.2-9 Core Imbalance Vs Power Level
Rancho Seco Cycle 5
0 - 60 EFPD
Imbalance Envelope

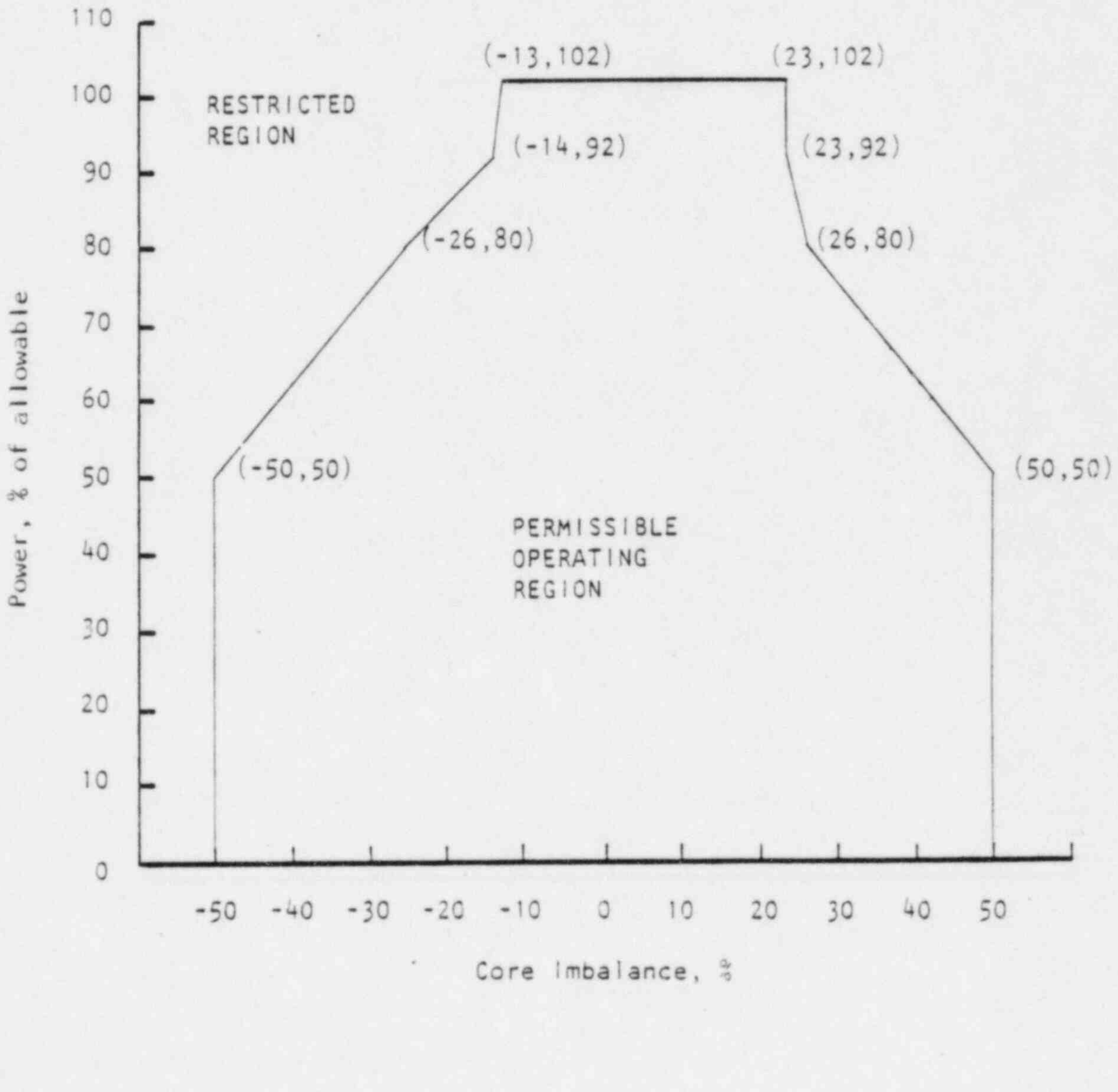


FIGURE 3.5.2-10 Core Imbalance Vs Power Level
Rancho Seco Cycle 5
50 - 250 EFPD (APSR In)
Imbalance Envelope

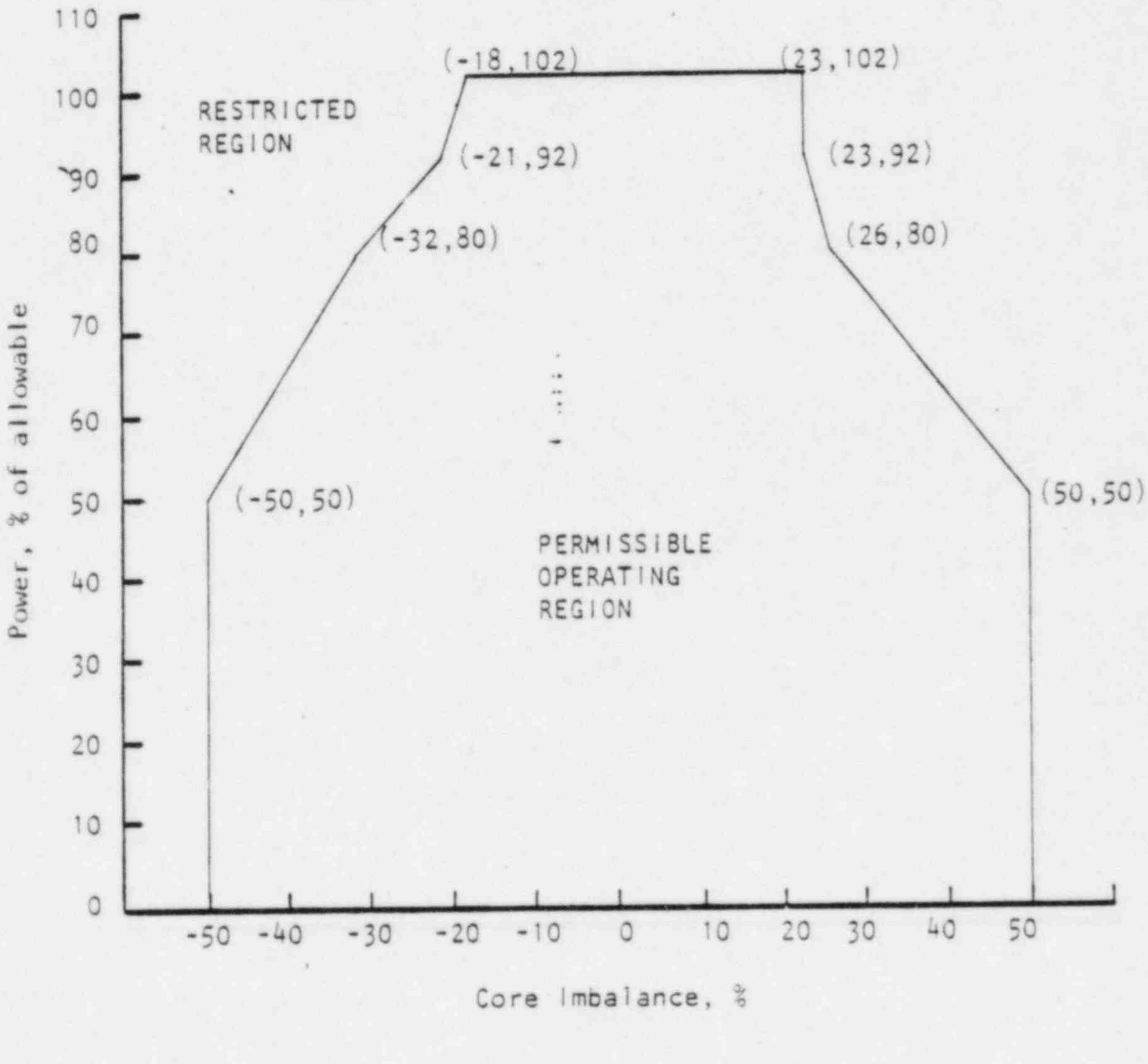


FIGURE 3.5.2-11 Core Imbalance Vs Power Level
Rancho Seco Cycle 5
230 EFPD - EOC (APSR Out)
Imbalance Envelope

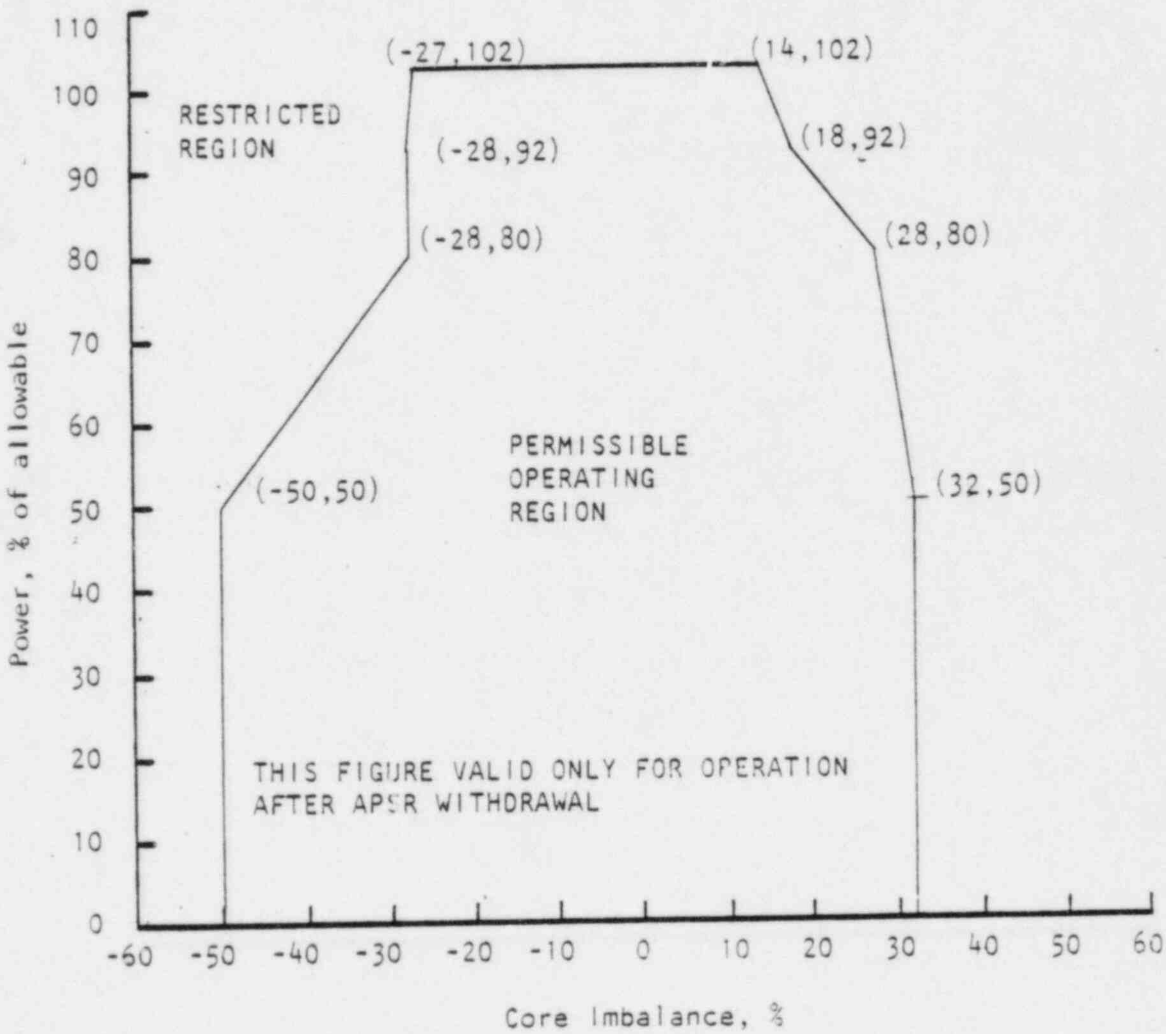
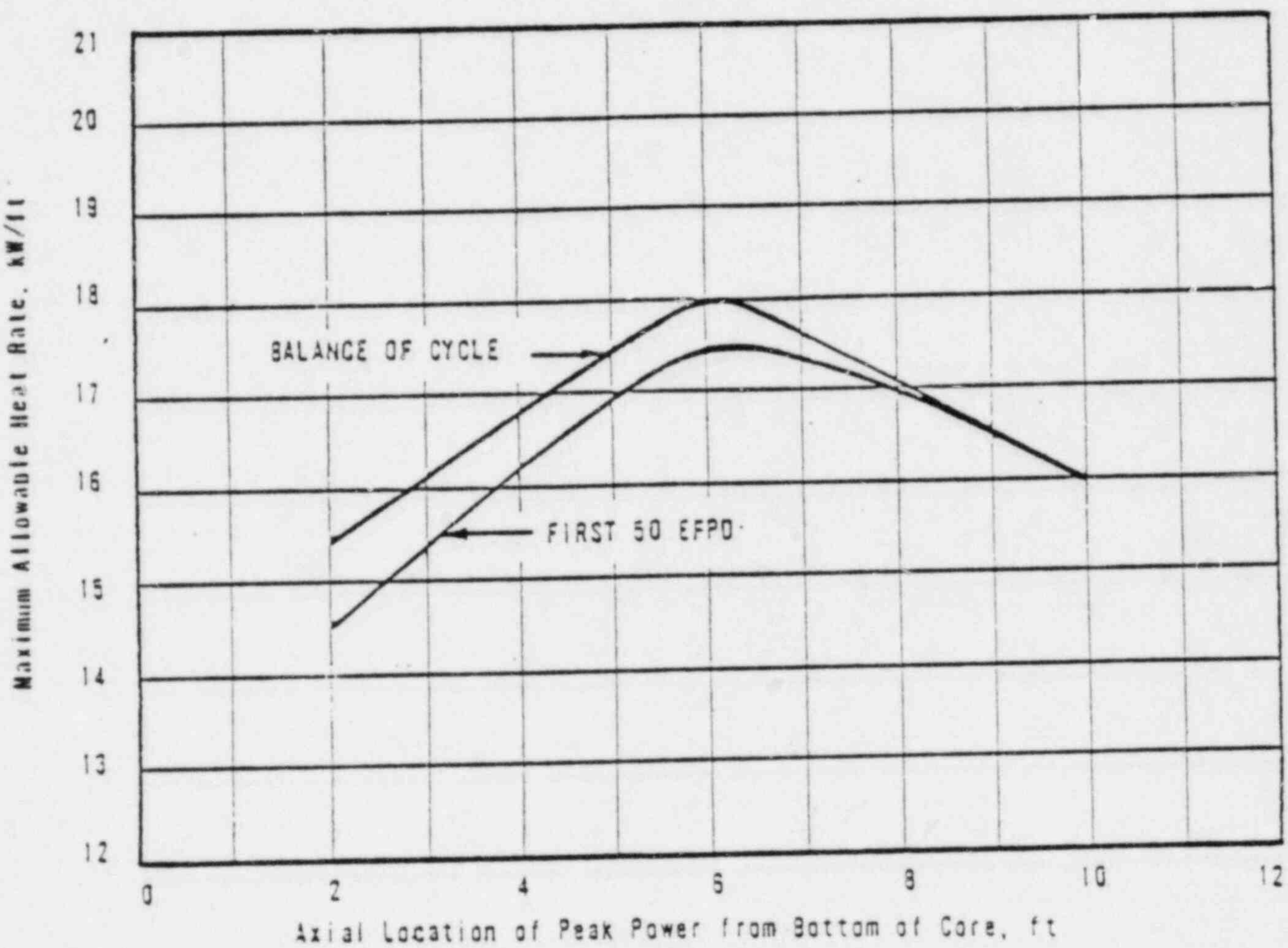


Figure 3.5.2-12

LOCA Limited Maximum Allowable Linear Heat Rate



3.5.3 Safety Features Actuation System Setpoints

Applicability

This specification applies to the safety features actuation system actuation setpoints.

Objective

To provide for automatic initiation of the safety features actuation system in the event of a breach of reactor coolant system integrity.

Specification

The safety features actuation setpoints and permissible bypasses shall be as follows:

FUNCTIONAL UNIT	ACTION	SETPOINT
High Reactor Building pressure*	Reactor Building spray valves***	$\leq$ 30 psig
	Reactor Building spray pumps***	$\leq$ 30 psig
	High pressure injection and start of Reactor Building cooling and Reactor Building isolation.	$\leq$ 4 psig
	Low pressure injection	$\leq$ 4 psig
Low reactor coolant system pressure**	High pressure injection and start of Reactor Building cooling and Reactor Building Isolation	$\geq$1600 psig
	Low pressure injection	$\geq$1600 psig
Automatic Actuation Logic	All above	Not applicable

3.5.3 Specification (Continued)

FUNCTIONAL UNIT	ACTION	SETPOINT
Manual	All above	Not applicable
Loss of all RC Pumps	Starts Auxiliary Feedwater Pumps	Not applicable
Low Feedwater Pressure	Starts Auxiliary Feedwater Pumps	$\geq$ 750 psig

*May be bypassed during Reactor Building leak rate test.

**May be bypassed below 1850 psig and is automatically reinstated above 1850 psig.

***Five-minute time delay.

With an SFAS setpoint less conservative than the values shown in the above table, declare the channel inoperable and apply the applicable operator action requirement (Column C) of Table 3.5.1-1.

Bases

High Reactor Building Pressure

The basis for the 30 psig and 4 psig setpoints for the high pressure signal is to establish a setting which would be reached in adequate time 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.

Low Reactor Coolant System Pressure

The basis for the 1600 psig low reactor coolant pressure setpoint for high and low pressure injection initiation is to establish a value which is high enough such that protection is provided for the entire spectrum of break sizes and is far enough below normal operating pressure to prevent spurious initiation.(1)

REFERENCES

(1) FSAR, paragraph 14.2.2.5

3.5.4 Incore Instrumentation

Applicability

Applies to the operability of the incore instrumentation system.

Objective

To specify the functional and operational requirements of the incore instrumentation system.

Specification

Above 80 percent of operating power determined by the reactor coolant pump combination, reference table 2.3.1, at least 23 individual incore detectors shall be operable to assist in the periodic calibration of the out-of-core detectors in regard to the core imbalance trip limits. The detectors shall be arranged as follows and may be a part of both basic arrangements.

3.5.4.1 Axial Imbalance

A. Three detectors in each of 3 strings shall lie in the same axial plane with 1 plane in each axial core half.

B. The axial planes in each core half shall be symmetrical about the core mid-plane.

C. The detector shall not have radial symmetry.

3.5.4.2 Radial Tilt

A. Two sets of 4 detectors shall lie in each core half. Each set of 4 shall lie in the same axial plane. The two sets in the same core half may lie in the same axial plane.

B. Detectors in the same plane shall have quarter core radial symmetry.

Bases

A system of 52 incore flux detector assemblies with 7 detectors per assembly has been provided primarily for fuel management purposes. The system includes data display and record functions and is also used for out-of-core instrumentation calibration and for core power distribution verification.

3.5.4 (Continued)

Bases (Continued)

A. The out-of-core nuclear instrumentation calibration includes:

1. Calibration of the split detectors at initial reactor startup, during the power escalation program, and monthly thereafter.

2. A comparison check with the incore instrumentation in the event one of the four out-of-core power range detector assemblies gives abnormal readings during power operation.

3. Confirmation that the out-of-core axial power splits are as expected.

B. Core power distribution verification includes:

1. Measurement at low power initial reactor startup to check that power distribution is consistent with calculations.

2. Subsequent checks during operation each 4,000 MWD/MTU average burnup to insure that power distribution is consistent with calculations.

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

C. The safety of unit operation at or below 80 percent of operating power(1) for the reactor coolant pump combinations without the core imbalance trip system has been determined by extensive 3-D calculations. This will be verified during the physics startup testing program.

D. The minimum requirement for 23 individual incore detectors is based on the following:

1. An adequate axial imbalance indication can be obtained with 9 individual detectors. Figure 3.5.4-1 shows a typical set of three detector strings with 3 detectors per string that will indicate an axial imbalance that is within 8 percent (calculated) of the real core imbalance. The three detector strings are the center one, one from the inner ring of symmetrical strings and one from the outer ring of symmetrical strings.

2. Figure 3.5.4-2 shows a typical detection scheme which will indicate the radial power distribution with 16 individual detectors. The readings from 2 detectors in a radial quadrant at either plane can be compared with readings from the other quadrants to measure a radial flux tilt.

3.5.4 (Continued)

Bases (Continued)

3. Figure 3.5.4-3 combines figures 3.5.4-1 and 3.5.4-2 to illustrate a typical set of 23 individual detectors that can be specified as a minimum for axial imbalance determination and radial tilt indication, as well as for the determination of gross core power distributions. Startup testing will verify the adequacy of this set of detectors for the above functions.

E. At least 23 specified incore detectors will be operable to check power distribution above 80 percent power determined by reactor coolant pump combination. These incore detectors will be read out either on the computer or on a recorder. If a set of 23 detectors in specified locations is not operable, power will be decreased to or below 80 percent for the operating reactor coolant pump combination.

REFERENCE

(1) FSAR, paragraph 7.1.2.2.3

FIGURE 3.5.4-1
INCORE INSTRUMENTATION.SPECIFICATION
AXIAL IMBALANCE INDICATION

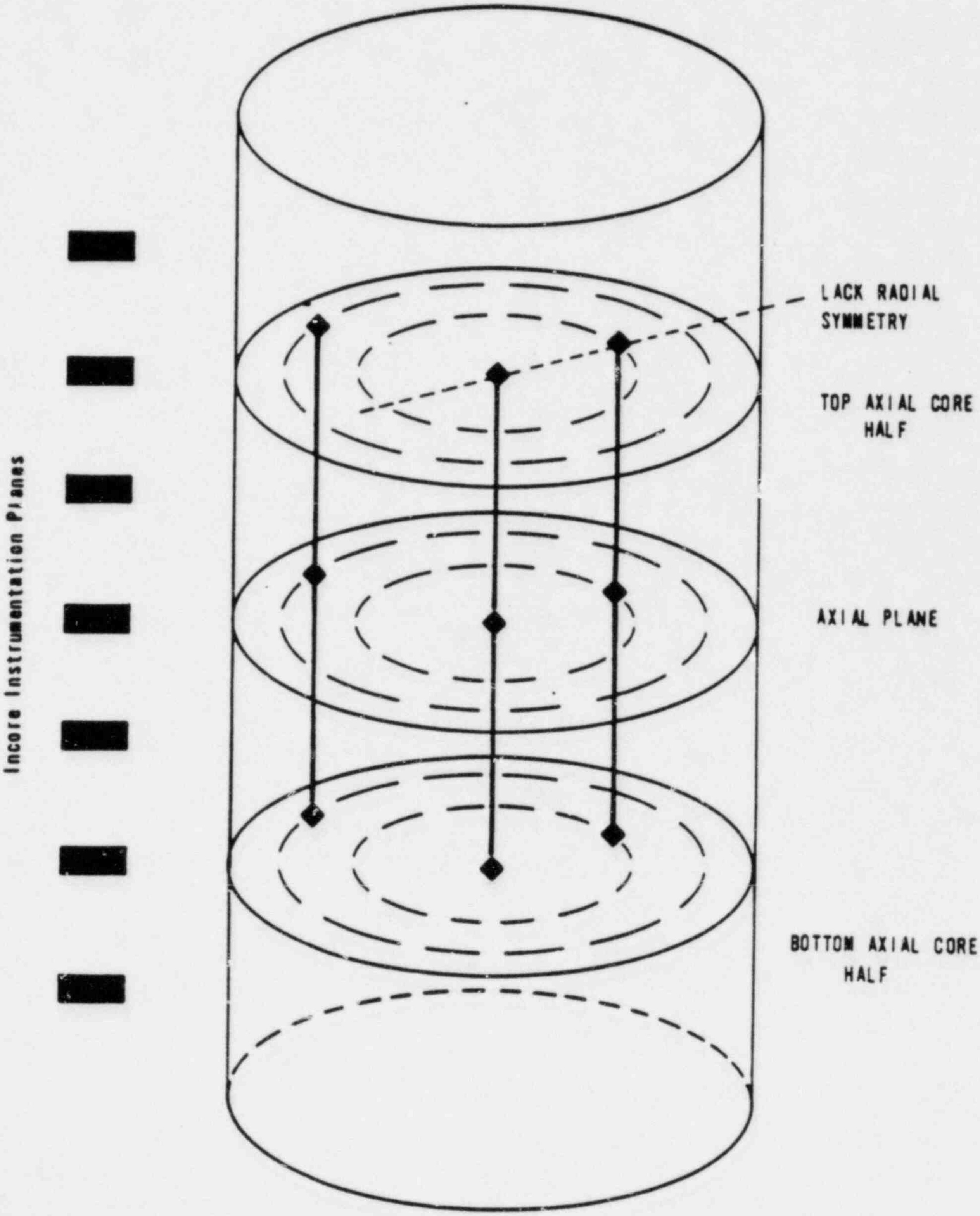


FIGURE 3.5.4-2
INCORE INSTRUMENTATION SPECIFICATION
RADIAL FLUX TILT INDICATION

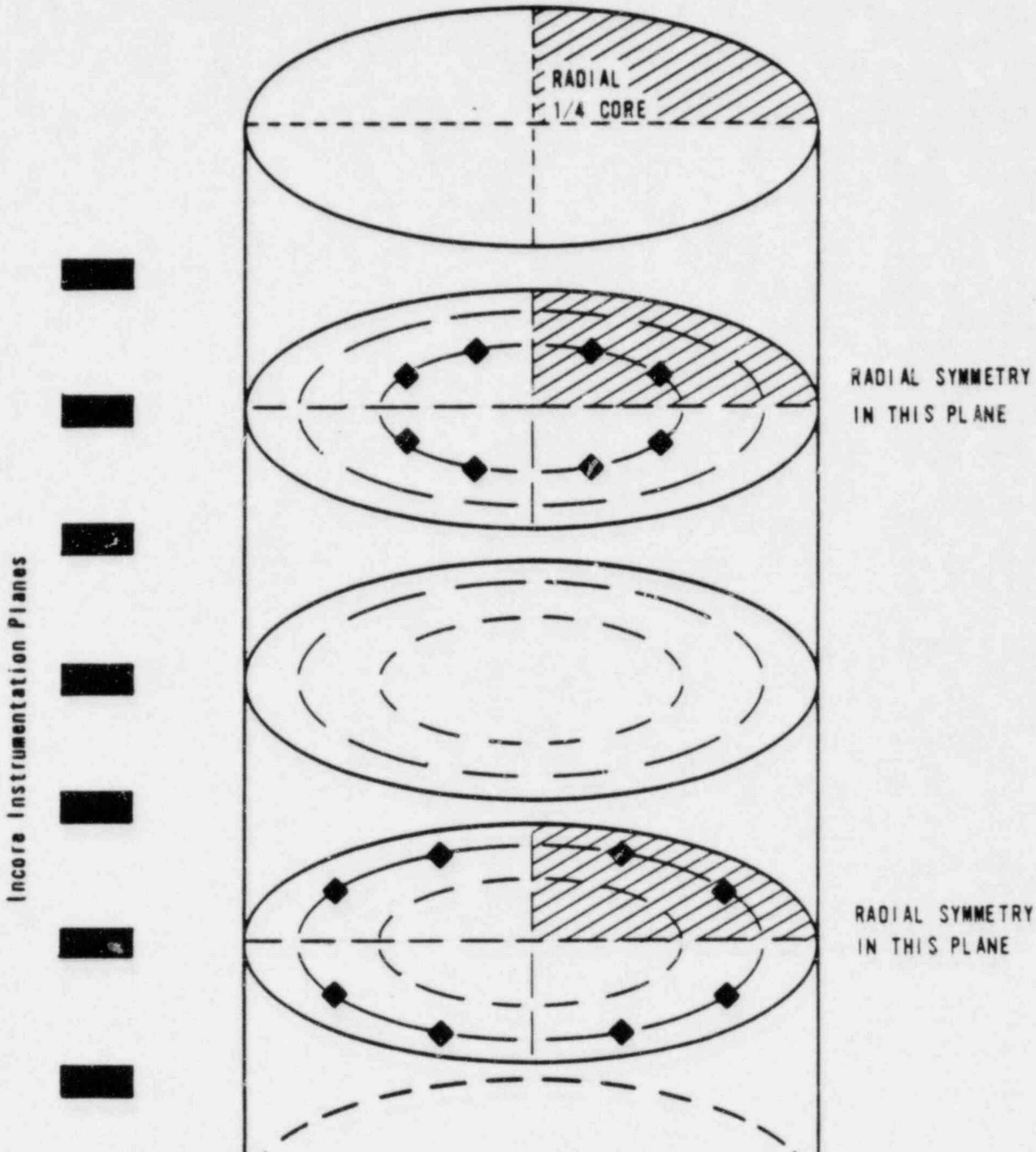
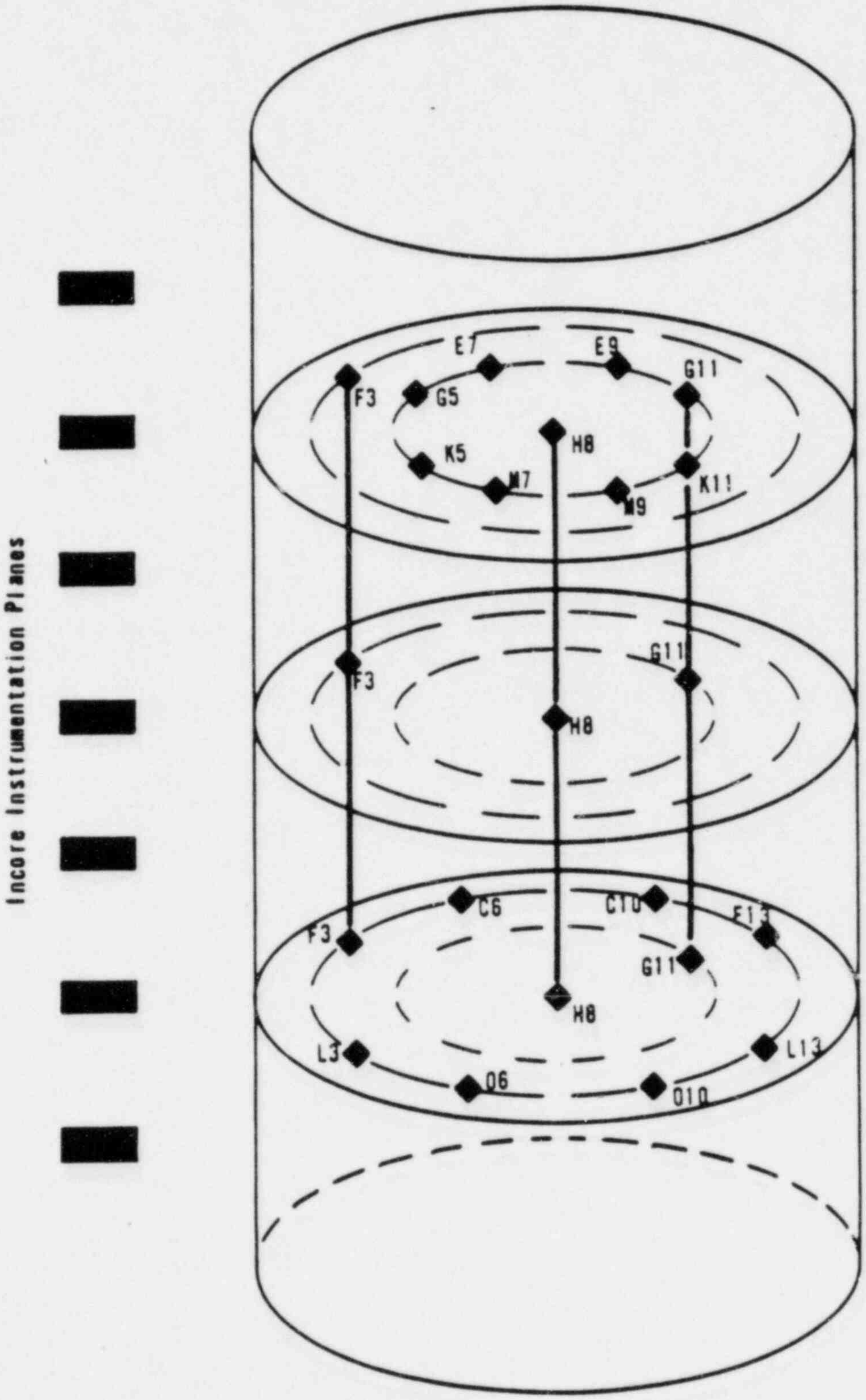


FIGURE 3.5.4-3
INCORE INSTRUMENTATION SPECIFICATION



3.6 REACTOR BUILDING

Applicability

Applies to the containment when the reactor is subcritical by less than 1 percent Δ k/k.

Objective

To assure containment integrity during startup and operation.

Specification

3.6.1 Containment integrity shall be maintained whenever all three of the following conditions exist:

A. Reactor coolant pressure is 300 psig or greater.

B. Reactor coolant temperature is 200 F or greater.

C. Nuclear fuel is in the core.

3.6.2 Containment integrity shall be maintained when the reactor coolant system is open to the containment atmosphere and the requirements for a refueling shutdown are not met.

3.6.3 Positive reactivity insertions which would result in the reactor being subcritical by less than 1 percent Δ k/k shall not be made by control rod motion or boron dilution whenever the containment integrity is not intact.

3.6.4 The reactor shall not remain critical if the Reactor Building internal pressure exceeds 1.5 psig or vacuum exceeds -1.5 psig.

3.6.5 Prior to criticality following refueling shutdown, a check shall be made to confirm that all manual containment isolation valves which should be closed are closed.

3.6.6 The safety features containment isolation valves specified in Table 3.6-1 shall be OPERABLE with closure times as shown in Table 3.5-1. If, under critical operations, an automatic containment isolation valve is determined to be inoperable, the other containment isolation valve in the line shall be tested to insure operability. If the inoperable valve is not restored within 48 hours, the reactor shall be brought to the cold shutdown condition within an additional 24 hours or the valve will be placed in a safety features position.

Table 3.6-1

SAFETY FEATURES CONTAINMENT ISOLATION VALVES

VALVE NUMBER	DESCRIPTION	MAXIMUM CLOSURE TIME (SEC)
SFV 53612	RB Atm. & Purge Sample, AB Side	3
SFV 53613	RB Atm. & Rad Sample, AB Side	3
SFV 60003	RC Sys. Drain Isol., AB Side	14
SFV 66308	RB Normal Sump Drain, AB Side	15
SFV 92520	Przr. Nitrogen Isol., AB Side	5
SFV 53503	RB Purge Inlet, AB Side	5
SFV 53604	RB Purge Outlet, AB Side	3
SFV 53610	RB Press. Equalizer, AB Side	15
SFV 60002	RC System Vent Isol., AB Side	6
SFV 60004	RC System Drain Isol., AB Side	14
SFV 66309	RB Normal Sump Drain, AB Side	8
SFV 70002	Pzrz. Liquid Sample Isol., AB Side	8
SFV 72502	Pzrz. Gas Sample Isol., AB Side	5
HV 20611	OTSG's Blowdown Isol., AB Side	22
HV 20593	OTSG-A Sample Isol., AB Side	12
HV 20594	OTSG-B Sample Isol., AB Side	5
SFV 53504	RB Purge Inlet, RB Side	8
SFV 53603	RB Press. Equalizer, RB Side	9
SFV 53605	RB Purge Outlet, RB Side	8
SFV 60001	RC Sys. Vent Isol., RB Side	12
SFV 70001	Pzrz. Liquid Sample Isol., RB Side	18
SFV 70003	Przr. Vapor Sample Isol., RB Side	21
SFV 72501	Przr. Gas Samale Isol., RB Side	9
*SFV 46014	RB CCW Supply, AB Side	15
*SFV 46203	RB CCW Return, RB Side	14
*SFV 46204	RB CCW Return, AB Side	18
*SFV 46906	CRD Cooling Water Supply, AB Side	9
*SFV 46907	CRD Cooling Water Return, RB Side	14
*SFV 46908	CRD Cooling Water Return, AB Side	8
*HV 20609	OTSG-A Blowdown Isol., RB Side	15
*HV 20610	OTSG-B Blowdown Isol., RB Side	14
SFV 22023	RC Sys. Letdown, RB Side	15
SFV 22009	RC Sys. Letdown, AB Side	6
SFV 24004	RC Pump Seal Return, RB Side	71
SFV 24013	RC Pump Seal Return, AB Side	8

*Manual initiation signal (no auto. initiation).

3.6 (Continued)

Bases

The reactor coolant system conditions of cold shutdown assure that no steam will be formed and hence no pressure buildup in the containment if the reactor coolant system ruptures.

The selected shutdown conditions are based on the type of activities that are being carried out and will preclude criticality in any occurrence.

The Reactor Building is designed for an internal pressure of 59 psig and an external pressure 2.0 psi greater than the internal pressure. The design external pressure corresponds to the differential pressure that could be developed if the building is sealed with an internal temperature of 120 F with a barometric pressure of 29.0 inches of Hg and the building is subsequently cooled to an internal temperature of 80 F with a concurrent rise in barometric pressure to 31.0 inches of Hg.

When containment integrity is established, the limits of 10 CFR 100 will not be exceeded should the maximum hypothetical accident occur.

The OPERABILITY of the containment isolation ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere by pressurization of the containment. Containment isolation within the time limits specified ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for LOCA.

REFERENCES

FSAR, section 5

3.7 AUXILIARY ELECTRICAL SYSTEMS

Applicability

Applies to the availability of off-site and on-site electrical power for station operation and for operation of station auxiliaries.

Objective

To define those conditions of electrical power availability necessary to provide for safe reactor operation and to provide for continuing availability of engineered safety features systems in an unrestricted manner.

Specification

3.7.1 The reactor shall not be brought critical unless the following conditions are met:

A. All nuclear service buses, nuclear service switchgear, and nuclear service load shedding systems are operable.

B. Two 220 kV lines are in service.

C. One 6900 volt reactor coolant pump motors bus is energized.

D. Emergency diesel generators are operable and at least 35,000 gallons of fuel are in each storage tank.

E. Plant batteries are charged and in service.

F. Two out of three battery chargers are operable for 125 volt d-c buses "A" and "C", and "B" and "D".

G. Three out of four inverters are operable for 120 volt a-c vital bus power.

H. Both startup transformers, No. 1 and No. 2, are in service.

3.7.2 The reactor shall not remain critical unless all of the following requirements are satisfied:

A. One 220 kV line shall be fully operational and capable of carrying nuclear service and auxiliary power except as specified in D below.

3.7.2 (Continued)

B. Both startup transformers shall be in service except that one will be sufficient if during the time one startup transformer is inoperable, a diesel generator is started and run continuously.

C. Both diesel generators shall be operable except that from and after the date that one of the diesel generators is made or found to be inoperable for any reason, reactor operation is permissible for the succeeding 15 days provided that during such 15 days the operable diesel generator shall be load tested daily and both startup transformers are available. If the diesel is not returned to service at the end of 15 days, the other diesel will be started and run with at least minimum load continously for an additional 15 days. If at the end of the second 15 days the diesel is not returned to service, the reactor shall be brought to the cold shutdown within an additional 24 hours.

D. If the plant is separated from the system while carrying its own auxiliaries, or if all 220 kV lines are lost, continued reactor operation is permissible provided that one emergency diesel generator shall be started and run continously until a transmission line is restored.

E. The essential nuclear service electrical buses, switchgear, load shedding, and automatic diesel start systems shall be operable except as provided in C above and as required for surveillance testing.

F. Nuclear service batteries are charged and in service except that one nuclear service battery may be removed from service for not more than 24 hours.

G. Both nuclear services buses are operable except that one nuclear service bus may be removed from service for not more than 24 hours provided that all equipment on the other nuclear service bus is operable.

3.7.3 If both diesel generators become inoperable, the unit shall be placed in the cold shutdown condition.

3.7.4 The pressurizer shall be OPERABLE with at least 126 kw of pressurizer heaters. With the pressurizer inoperable due to inoperable emergency power supplies to the pressurizer heater either restore the inoperable emergency power supply within 72 hours or be in at least HOT STANDBY within the next six hours and in HOT SHUTDOWN within the following 12 hours.

3.7 (Continued)

Bases

The auxiliary electrical power systems are arranged so that no single failure can inactivate enough safety features equipment to jeopardize plant safety.

The normal source of power to the redundant nuclear service loads is by the two startup transformers connected to the 220-kV station switchyard. All of the normal power supply to plant auxiliary loads is provided through the two unit auxiliary transformers connected to the generator bus. Emergency power for the nuclear service loads is obtained from two on-site diesel generators. Since the startup transformers are sized to carry full plant auxiliary loads, if plant auxiliaries' power is not available from the unit auxiliary transformer it will be obtained from the startup transformers.

The five 220-kV transmission lines are not under the direct control of the Rancho Seco station. Therefore, all five cannot be assumed to be available at all times. However, extensive reliability and protective features are utilized so that the probability of losing more than one source of 220-kV power from faults is low. By requiring that two 220-kV lines are in service prior to startup, one circuit will be immediately available following a loss of the onsite alternating current diesel power supplies and the other offsite 220-kV line. If there is a loss of all 220-kV remote connections, power to the safety features will be supplied by the diesel generators.

The 35,000 gallons of fuel stored in each storage tank permit operation of the two diesel generators for seven days. It is considered unlikely not to be able to secure fuel oil from an outside source during this time under the worst of weather conditions.

The four 125-volt d-c control panelboards are arranged so that loss of one bus will not preclude safe shutdown or operation of safety features systems. During periods when one plant battery is de-energized for test or maintenance, the associated 125-volt d-c bus can be supplied from its battery charger.

Each redundant pair ("A" and "C", "B" and "D") of safety features actuation and reactor protection 125-volt d-c buses has a standby battery charger in addition to the two bus battery chargers. Loss of power from one battery charger per pair of redundant d-c buses has no significant consequence since a standby battery charger is available. In addition, each 125-volt d-c bus can continue to receive power from its respective battery without interruption.

Sufficent redundancy is available with any three of the four 120-volt a-c vital power buses in service that reactor safety is assured. Every reasonable effort will be made to maintain all safety instrumentation in operation.

3.7 (Continued)

Bases (Continued)

During periods of station operation under the condition of electrical system degradation, as described above in Specification 3.7.2, the operating action required is to start and run sufficient standby power supplies so as not to compromise the safety of the plant. As seen in Specification 3.7.2, a time limit is placed on operation during certain degraded conditions based on the reliability of the available power supply.

The requirement that 126 kw of pressurizer heaters and their associated controls being capable of being supplied with electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation at HOT SHUTDOWN.

REFERENCE

FSAR, section 8

3.8 FUEL LOADING AND REFUELING

Applicability

Applies to fuel loading and refueling operations.

Objective

To assure that fuel loading and refueling operations are performed in a responsible manner.

Specification

3.8.1 Radiation levels in the Reactor Building refueling area shall be monitored by R15026 and R15027. Radiation levels in the spent fuel storage area shall be monitored by R15028. If any of these instruments become inoperable, portable survey instrumentation, having the appropriate ranges and sensitivity to fully protect individuals involved in refueling operation, shall be used until the permanent instrumentation is returned to service.

3.8.2 Core subcritical neutron flux shall be continously monitored by at least two neutron flux monitors, each with continous indication available, whenever core geometry is being changed. When core geometry is not being changed, at least one neutron flux monitor shall be in serivce.

3.8.3 At least one decay heat removal pump and cooler shall be operable.

3.8.4 During reactor vessel head removal and while loading and unloading fuel from the reactor, the boron concentration shall be maintained at not less than that required for refueling shutdown.

3.8.5 Direct communications between the control room and the refueling personnel in the Reactor Building shall exist whenever changes in core geometry are taking place.

3.8.6 During the handling of irradiated fuel in the Reactor Building at least one door on the personnel and emergency hatches shall be closed. The equipment hatch cover shall be in place with a minimum of four bolts securing the cover to the sealing surfaces.

3.8.7 Isolation valves in lines containing automatic containment isolation valves shall be operable, or at least one shall be in a safety features position.

3.8 Specification (Continued)

3.8.8 When two irradiated fuel assemblies are being handled simultaneously within the fuel transfer canal, a minimum of 10 feet separation shall be maintained between the assemblies at all times. Irradiated fuel assemblies may be handled with the auxiliary bridge crane provided no other irradiated fuel assembly is being handled in the fuel transfer canal.

3.8.9 If any of the above specified limiting conditions for fuel loading and refueling are not met, movement of fuel into the reactor core shall cease; action shall be initiated to correct the conditions so that the specified limits are met, and no operations which may increase the reactivity of the core shall be made.

3.8.10 The Reactor Building purge system, including the radiation monitors, R15001A and R15001B, shall be tested and verified to be operable immediately prior to refueling operations.

3.8.11 Irradiated fuel shall not be removed from the reactor until the unit has been subcritical for at least 72 hours.

3.8.12 No loads will be handled over irradiated fuel stored in the spent fuel pool, except the fuel assemblies themselves. A dead weight load test at the rated load will be performed on the Fuel Storage Building handling bridge prior to each refueling.

Bases

Detailed written procedures will be available for use by refueling personnel. These procedures, the above specifications, and the design of the fuel handling equipment are described in subsection 9.7 of the FSAR 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. If no change is made in core geometry, one flux monitor is sufficient. This permits maintenance on the instrumentation. Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition. The decay heat removal pump is used to maintain a uniform boron concentration. (1) The refueling boron concentration indicated in Specification 3.8.4 will be maintained to ensure that the more restrictive of the following reactivity conditions is met:

1. Either a k_{eff} of 0.95 or less with all control rods removed from the core.

2. A boron concentration of $\geq$ 1800 ppm.

Specification 3.8.5 allows the control room operator to inform the Reactor Building personnel of any impending unsafe condition detected from the main control board indicators during fuel movement.

3.8 (Continued)

Bases (Continued)

The specification requiring testing Reactor Building purge termination is to verify that these components will function as required should a fuel handling accident occur which resulted in the release of significant fission products.

Specification 3.8.11 is required as the safety analysis for the fuel handling accident was based on the assumption that the reactor had been shutdown for 72 hours and all 208 fuel pins in the hottest fuel assembly fail releasing all gap activity. (2)

REFERENCES

(1) FSAR, subsection 9.5

(2) FSAR, paragraph 14.2.2.3.2

3.9 Not Used

3.10 SECONDARY SYSTEM ACTIVITY

Applicability

Applies to the limiting conditions of secondary system activity for operation of the reactor.

Objective

To limit the maximum secondary system activity.

Specification

The reactor shall not remain critical if the iodine 131 activity in the secondary side of a steam generator exceeds 0.2 μ Ci/cc.

Bases

For the purpose of determining a maximum allowable secondary coolant activity, the activity contained in the mass released following a loss of load accident is considered. As stated in FSAR paragraph 14.1.2.8.3, 224,000 pounds of water are released to the atmosphere via the relief valves. A site boundary dose limit of 1.5 rem is used. This is the recommended annual dose limit to the thyroid for general population. (1)

The whole body dose is negligible since any noble gases entering the secondary coolant system are continuously vented to the atmosphere by the condenser air ejector, thus, in the event of a loss of load incident there are only small quantities of these gases which would be released.

I^{131} is the significant isotope because of its low MPC in air and because the other iodine isotopes have shorter half-lives, and therefore, cannot build up to significant concentrations in the secondary coolant, given the limitations on primary leak rate and technical specification limiting activity. One-tenth of the contained iodine is assumed to reach the site boundary, making allowance for plateout and retention in water droplets. I^{131} is assumed to contribute 70 percent of the total thyroid dose based on the ratio of I^{131} to the total iodine isotopes given in Table 11-3 of the FSAR.

The maximum inhalation dose at the site boundary is then as follows:

$$\text{Dose (rem)} = Ci \cdot V \cdot B \cdot DCF \cdot (0.1) \cdot X/Q$$

Ci = secondary coolant activity (0.286 μ Ci/cc I^{131} equivalent)

V = secondary water volume released to atmosphere (102 m^3)

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

3.10 (Continued)

Bases (Continued)

X/Q = ground level release dispersion factor (8.51×10^{-4} sec/m^3)

DCF = 1.48×10^6 rem/Ci

0.1 = fraction of activity released

The resultant dose is 1.28 rem compared to the limit of 1.5 rem.

REFERENCES

(1) Background Material for the Development of Radiation Standards, Report No. 2, Federal Radiation Council, September 1961.

3.11 REACTOR BUILDING POLAR CRANE AND AUXILIARY HOIST

Applicability

Applies to the use of the Reactor Building polar crane over the steam generator compartments and the fuel transfer canal and the auxiliary hoist over the fuel transfer canal.

Objective

To identify those conditions for which the operation of the Reactor Building polar crane and auxiliary hoist are restricted.

Specification

3.11.1 The Reactor Building polar crane hoists shall not be operated over the fuel transfer canal when any fuel assembly is being moved.

3.11.2 The auxiliary hoist shall not be operated over the fuel transfer canal when any fuel assembly is being moved.

3.11.3 During the period when the reactor vessel head is removed and irradiated fuel is in the Reactor Building and fuel is not being moved, the Reactor Building polar crane and auxiliary hoist shall be operated over the fuel transfer canal only where necessary and in accordance with approved operating procedures stating the purpose of such use. At no time will heavy loads be transported over irradiated fuel, except those components which must be moved during refueling.

3.11.4 During the period when the reactor coolant system is pressurized above 300 psig, and is above 200 F, and fuel is in the core, the Reactor Building polar crane shall not be operated over the steam generator compartments.

Bases

Restriction of use of the Reactor Building polar crane and auxiliary hoist over the fuel transfer canal when the reactor vessel head is removed to those operations necessary for the fuel handling and core internals operations is to preclude the dropping of materials or equipment into the reactor vessel and possibly damaging the fuel to the extent that an escape of fission products would result. The fuel transfer canal will be delineated by readily visible markers at an elevation above which the Reactor Building polar crane would not normally handle loads.

Restriction of use of the Reactor Building polar crane over the steam generator compartments during the time when steam could be formed from dropping a load on the steam generator or reactor coolant piping resulting in rupture of the system is required to protect against a loss of coolant accident.

3.12 SHOCK SUPPRESSORS (SNUBBERS)

Applicability

This specification applies to the operability of safety systems shock suppressors (snubbers) during power operation.

Objective

To assure that safety related snubbers are operable to protect the nuclear system during seismic transients when the reactor is operating.

Specification

1. During power operation all safety-related snubbers listed in Table 4.14-1 shall be operable, except as noted in 3.12.2 through 3.12.4 below.

2. If the requirements of 3.12.1 cannot be met, maintenance shall be allowed during power operation on any safety related snubber listed on Table 4.14.1. If the snubber being repaired is not restored to operability or replaced within 72 hours, the reactor shall be placed in a hot shutdown condition within 12 hours. If operability or replacement is not accomplished within an additional 48 hours, the reactor shall be placed in a cold shutdown condition within 24 hours.

3. If a snubber is determined to be inoperable while the reactor is in the cold shutdown, hot shutdown or refueling shutdown mode, the snubber shall be made operable or replaced prior to reactor startup.

4. Snubbers may be added to safety related systems without prior NRC License Amendments to Table 4.14-1 provided that a revision to this table is included with the subsequent License Amendment request.

Bases

Snubbers are designed to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient, while allowing normal thermal motion during startup and shutdown. The consequence of an inoperable snubber is an increase in the probability of structural damage to piping as a result of a seismic or other event initiating dynamic loads. It is therefore required that all snubbers required to protect the primary coolant system or any other safety system or component be operable within a limited time interval during reactor operation.

3.13 Reserved

3.14 FIRE SUPPRESSION

3.14.1 Instrumentation

Specification

3.14.1.1 As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 3.14-1 shall be OPERABLE.

3.14.1.2 With the number of instruments OPERABLE less than required by the minimum instruments OPERABLE requirement of Table 3.14-1:

a. Within one (1) hour, establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect the containment at least once per 8 hours or monitor the containment air temperature at least once per hour from recorder TJR-07 at the points listed below:

Point 01 Reactor Building Cooling Units A & B Air In
Point 02 Reactor Building Cooling Units C & D Air In
Point 12 Top of Reactor Shielding
Point 13 Top of Reactor Shielding
Point 16 Top of Reactor Building

b. Restore the inoperable instrument (s) to OPERABLE status within fourteen (14) days; or

c. In lieu of any other report required by Specification 6.9, prepare and submit a special report to the Commission, pursuant to Specification 6.9.5.E, within the next thirty (30) days outlining the action taken, the cause of inoperability and the plans and schedule for restoring the instrument (s) to OPERABLE status.

3.14.2 Water System

Specification

3.14.2.1 The FIRE SUPPRESSION WATER SYSTEM shall be OPERABLE with:

a. Two high pressure pumps each with a capacity of 2000 gal/min with their discharge aligned to the fire suppression header.

b. Two separate water supplies containing a minimum of 2,000,000 gallons each.

3.14.2.1 (Continued)

c. An OPERABLE flow path capable of taking suction from the circulating water system and the Site Reservoir or the Folsom South Canal and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrant curb valves, the last valve ahead of the water flow alarm device on each sprinkler or hose standpipe, and the last valve ahead of the deluge valve on each deluge or spray system required to be OPERABLE per Specifications 3.14.3.1 and 3.14.5.

3.14.2.2 With one pump and/or one water supply inoperable, restore the inoperable equipment to OPERABLE status within 7 days or, in lieu of any other report required by Specification 6.9.5.E, prepare and submit a Special Report to the Commission pursuant to Specification 6.9 within the next thirty (30) days outlining the plans and procedures to be used to restore the inoperable equipment to OPERABLE status or to provide an alternate backup pump or supply.

3.14.2.3 With no fire suppression water system operable:

a. Establish a backup fire suppression water system equivalent to Specification 3.14.2.1 within 24 hours.

b. In lieu of any other report required by Specification 6.9, submit a Special Report in accordance with Specification 6.9.5.E:

   1) By telephone within 24 hours,

   2) Confirmed by telegraph, mailgram or facsimile transmission no later than the first working day following the event, and

c. In writing within 14 days following the event, outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

d. If a. above cannot be fulfilled, place the reactor in hot standby within the next six (6) hours and cold shutdown within the following thirty (30) hours.

TABLE 3.14-1

FIRE DETECTION INSTRUMENTS FOR SAFETY SYSTEMS

Zone	Instrument Location	Minimum Operable Heat	Minimum Operable Flame	Minimum Operable Smoke
4A	Control/Computer Room	0	0	8
4B	Control/Computer Room	0	0	15
11	Battery Room, Mezzanine Level	0	0	4
12	West DC Control Room, Mezzanine Level	0	0	2
13	West 480 VAC Room, Mezzanine Level	0	0	2
14	West Cable Tray Area	0	0	2
15	East Cable Tray Area	0	0	2
16	East 480 VAC Room, Mezzanine Level	0	0	2
17	East DC Control Room, Mezzanine Level	0	0	2
19	Communications Room	0	0	5
36	West Battery Room, Grade Level	0	0	4
37	West 4160 VAC Room	0	0	2
38	East 4160 VAC Room	0	0	2
39	East Battery Room	0	0	4
40	North Diesel Room	2	0	0
41	South Diesel Room	2	0	0
44	Reactor Coolant Pumps A and B	0	1	0
44	Reactor Coolant Pumps C and D	0	1	0
45	Electrical Penetration Area, Grade Level	0	0	2
47	HPI Pump A Room	0	0	1
47	West DH Cooler Room (valve penetration area)	0	0	3
47	East DH Cooler Room (Valve penetration area)	0	0	3
47	MU Pump Room	0	0	1
47	HPI Pump B Room	0	0	1
48	West DH Pump Room	0	0	1
48	East DH Pump Room	0	0	1

3.14.3 Spray and Sprinkler Systems

Specification

3.14.3.1 The spray and/or sprinkler systems located in the following areas shall be OPERABLE:

a. Control Room (Zone 3)
b. Controlled Area, Mezzanine Level (Zone 20)
c. Main Lube Oil Area, Grade Level (Zone 32)
d. Grade Level (Zone 34)
e. North Diesel Room (Zone 40)
f. South Diesel Room (Zone 41)
g. West Controlled Area, Grade Level (Zone 42)
h. East Controlled Area, Grade Level (Zone 43)
i. South and East -20' Level (Zone 46)

3.14.3.2 a. With one or more of the above, items a through f, required spray and/or sprinkler systems inoperable, within one hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components required to safely shut down and cool down the plant could be damaged; for other areas, establish an hourly fire watch patrol. Restore the system to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.5.E within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

3.14.4 CO_2 System

Specification

3.14.4.1 The CO_2 systems located in the following areas shall be OPERABLE with a minimum capacity of 66% and a minimum pressure of 275 psig in the storage tank.

a. Zone 12 West DC Control Room Mezzanine Level
b. Zone 13 West 480 VAC Room Mezzanine Level
c. Zone 14 West Cable Tray Area
d. Zone 15 East Cable Tray Area
e. Zone 16 East 480 VAC Room Mezzanine Level

3.14.4.1 (Continued)

f. Zone 17 East DC Control Room Mezzanine Level
g. Zone 36 West Battery Room Grade Level
h. Zone 37 West 4160 VAC Room
i. Zone 38 East 4160 VAC Room
j. Zone 39 East Battery Room
k. Zone 40 North Diesel Room
l. Zone 41 South Diesel Room
m. Zone 19 Communications Room

3.14.4.2 a. With one or more of the above required CO_2 systems inoperable, within one hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components required to safely shut down and cool down the plant could be damaged; for other areas, establish an hourly fire watch patrol. Restore the system to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.5.E within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

3.14.5 Fire Hose Stations

Specification

3.14.5.1 The fire hose stations in the following locations shall be OPERABLE:

a. All stations specified in Table 3.14-2

b. Hydrant #2

c. Hydrant #3

d. Hydrant at coordinates N59+17; E30+20

e. Hydrant at coordinates N59+17; E32+20

3.14.5.2 With one or more of the fire hose stations above inoperable, route an additional equivalent size fire hose to the unprotected area(s) from an OPERABLE hose station within 1 hour if the inoperable fire hose is the primary means of fire suppression; otherwise, route the additional hose within 24 hours. Restore the fire hose station to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.5.E within the next 30 days outlining the action taken, the cause of the inoperability, and plans and schedule for restoring the station to OPERABLE status.

3.14.6 Fire Barrier Penetration Fire Seals

Specification

3.14.6.1 All fire barrier penetrations (including cable penetration barriers, fire doors and fire dampers) in fire zone boundaries protecting safety related areas shall be functional.

3.14.6.2 With one or more of the above required fire barrier penetrations non-functional, within one hour either establish a continuous fire watch on at least one side of the affected penetration, or verify the OPERABILITY of fire detectors on at least one side of the non-functional fire barrier and establish an hourly fire watch patrol. Restore the non-functional fire barrier penetrations to functional status within 7 days or, in lieu of any other report required by Specification 6.9, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.5.E within the next 30 days outlining the action taken, the cause of the non-functional penetration and plans and schedule for restoring the fire barrier penetrations to functional status.

Bases

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety related equipment and is an integral element in the overall facility fire protecion program.

3.14 (Continued)

Bases (Continued)

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is returned to service.

The OPERABILITY of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety related equipment is located. The fire suppression system consists of the water system, spray and/or sprinklers, CO_2, and fire hose stations. The collective capability of the fire suppression systems is adequate to minimize potential damage to safety related equipment and is a major element in the facility fire protection program.

In the event that portions of the fire suppression systems are inoperable, alternate backup fire fighting equipment is required to be made available in the affected areas until the affected equipment can be restored to service.

In the event that the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant. The requirement for a twenty-four (24) hour report to the Commission provides for prompt evaluation of the acceptability of the corrective measures to provide adequate fire suppression capability for the continued protection of the nuclear plant.

The functional integrity of the fire barrier penetration seals ensures that fires will be confined or adequately retarded from spreading to adjacent portions of the facility. This design feature minimizes the possibility of a single fire rapidly involving several areas of the facility prior to detection and extinguishment. The fire barrier penetration seals are a passive element in the facility fire protection program and are subject to periodic inspections.

During periods of time when the seals are not functional, an hourly fire watch is required to be maintained in the vicinity of the affected seal until the seal is restored to functional status.

TABLE 3.14-2
AUXILIARY BUILDING FIRE HOSE STATIONS

ID No.	Location
HS 1	+40 ft. Level Corridor by Counting Room
HS 2	+40 ft. Level Corridor Across from Control Room
HS 3	+40 ft. Level Corridor by Chemistry Lab
HS 4	+40 ft. Level Corridor by Cleaning Room
HS 5	+20 ft. Level in Ventilation Equipment Room
HS 6	+20 ft. Level Corridor by Communications Room
HS 7	Grade Level by CRD Cooling Water Heat Exchanger
HS 8	+20 ft. Level Corridor by Ventilation Equipment Room
HS 9	+20 ft. Level Corridor by A/C Equipment Room
HS 10	Grade Level Corridor by Diesel Generator Room
HS 11	Grade Level Corridor Across from East 4160 Switchgear Room
HS 12	Grade Level Corridor Across from East Battery Room
HS 13	Grade Level Corridor by East End Stairwell
HS 14	Grade Level by Waste Solidification Area
HS 15	-20 ft. Level HPI Pump A Room
HS 16	-20 ft. Level Containment Penetration Valve Area
HS 17	-20 ft. Level Corridor by Waste Gas Decay Tank Room
HS 18	-20 ft. Level Corridor North of BA Evaporator Room
HS 19	-47 ft. Level Stairway by East and West DHR Pump Rooms
HS 20	-20 ft. Level Corridor East of HPI Pump B Room

3.15 Not Used

3.16 Not Used

3.17 Not Used

3.18 Not Used

3.19 RADIOACTIVE LIQUID EFFLUENT INSTRUMENTATION

The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.19-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.21 are not exceeded.

Applicability: During radioactive releases via the pathways identified in Table 3.19-1.

Action:

a. With a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint less conservative than a value which will ensure that the limits of Specification 3.21 are met, without delay suspend the release of radioactive liquid effluents monitored by the affected channel or declare the channel inoperable or change the setpoint so it is acceptably conservative.

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.19-1.

Bases

During normal operations all radioactive contaminated water from primary system leaks and drains are processed in a liquid radwaste system and recycled into the Reactor Coolant Makeup System or otherwise reused in the controlled areas of the plant. Only secondary system water is normally released from the plant. The secondary system water, if contaminated, would be released through the Regenerant Hold-Up Tank.

During periods of primary to secondary leakage, or when the sumps are contaminated, administrative controls require the turbine building sumps liquid effluent to be diverted to the Regenerant Hold-Up Tanks.

Under normal conditions the once through steam generators have no blow down. If a blow down is required during periods of primary to secondary leakage, all water will be retained and processed in the radwaste system or diverted to the Regenerant Hold-Up Tank.

Upon indication of radioactivity in the secondary system, radioactive liquid effluent instrumentation is required to monitor and control, as applicable, the releases of radioactive materials in liquid effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

TABLE 3.19-1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Number of Channels Operable	Action
1. Gross Radioactivity Monitors Providing Automatic Termination of Release		
a. Regenerant Hold-Up Tank Discharge Line Monitor	1	With the monitor inoperable effluent releases may be resumed for up to 14 days provided that prior to initiating a release: 1. At least two independent samples are analyzed in accordance with Specification 3.21. 2. A second member of the facility technical or operational staff will independently verify the release rate calculations and discharge valving. 3. Exert best efforts to return the instrument to OPERABLE status within 30 days and, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.
2. Flow Rate Measurement Devices		
a. Regenerant Hold-up Tank Discharge Line Monitor	1	With the flow rate measurement device inoperable, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours during actual releases. Pump performance curves generated in situ may be used to estimate flow.

TABLE 3.19-1 (Continued)

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Number of Channels Operable	Action
2. Flow Rate Measurement Devices (Continued)		
b. Waste Water Flow	1	With the flow rate measurement device inoperable, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours during actual releases.

3.20 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.20-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.22 are not exceeded.

Applicability During release via the pathways identified in Table 3.20-1.

Action

a. With a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint less conservative than a value which will ensure that the limits of Specification 3.22 are met, declare the channel inoperable.

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.20-1.

Bases

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases. The alarm/trip setpoints for these instruments shall be calculated in accordance with the methods in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements and General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

The Waste Gas Header Monitor monitors the Waste Gas Holdup System noble gas releases and will provide automatic termination of the release. However, it is located on the system header and monitors the noble gas prior to dilution in the Auxiliary Building ventilation system and passing through HEPA and charcoal filters. The Auxiliary Building Stack alarms and terminates the release automatically if it exceeds the limits. Therefore, as the Auxiliary Building Stack is the effluent release point and will perform the necessary Waste Gas Holdup System release termination, it is listed as the technical specification instrument.

The air ejector exhaust and gland seal exhaust also have individual noble gas monitors. These systems exhaust into the Auxiliary Building ventilation system. Therefore, as the Auxiliary Building Stack is the effluent release point and will alarm if either of these systems releases environmentally significant radioactive gases, it is used as the technical specification instrument.

Table 3.20-1

RADIOACTIVE GASES EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Number of Channels Operable	Action
1. Reactor Building Purge Vent		
a. Noble Gas Activity Monitor providing alarm and automatic termination of release.	1	With the monitor inoperable, effluent releases via this pathway may continue provided grab samples are taken at least once per 8 hours and are analyzed in accordance with Table 4.22-1 within 24 hours.
b. Iodine Sampler	1	With the collection device inoperable, effluent releases via this pathway may continue provided continous samples are taken and these samples are analyzed in accordance with Table 4.22-1 within 24 hours.
c. Particulate Monitor	1	With the collection device inoperable, effluent releases via this pathway may continue provided continous samples are taken and these samples are analyzed in accordance with Table 4.22-1 within 24 hours.
d. System Effluent Flow Rate Device	1	With the flow rate device inoperable, effluent releases may continue provided the flow rate used is the maximum design flow rate.
e. Sampler Flow Rate Measurement Device		With the flow rate device inoperable, effluent releases via this pathway may continue provided the flow rate is estimated and recorded at least once per 4 hours.

Table 3.20-1 (continued)

RADIOACTIVE GASES EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Number of Channels Operable	Action
2. Auxiliary Building Stack		
a. Noble Gas Activity Monitor providing alarm and automatic termination of release.	1	With the monitor inoperable, effluent releases via this pathway may continue provided grab samples are taken at least once per 8 hours and these samples are analyzed in accordance with Table 4.22-1 within 24 hours.
b. Iodine Sampler	1	With the collection device inoperable, effluent releases via this pathway may continue provided continuous samples are taken and these samples are analyzed in accordance with Table 4.22-1 within 24 hours.
c. Particulate Monitor	1	With the collection device inoperable, effluent releases via this pathway may continue provided continuous samples are taken and these samples are analyzed in accordance with Table 4.22-1 within 24 hours.
d. System Effluent Flow Rate Device	1	With the flow rate device inoperable, effluent releases via this pathway may continue provided the flow rate used is the maximum design flow rate.
e. Sampler Flow Rate Measuring Device	1	With the flow rate device inoperable, effluent releases via this pathway may continue provided the flow rate is estimated and recorded at least once per 4 hours.

Table 3.20-1 (continued)

'ADIOACTIVE GASES EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Number of Channels Operable	Action
2. Auxiliary Building Stack (continued)		
f. Waste Gas Holdup System (Auxiliary Building Stack Monitor)	1	With the monitor inoperable the contents of the tank(s) may be released to the environment provided that prior to initiating the release: a. At least two independent samples of the tank's contents are analyzed, and b. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge valve lineup; Otherwise, suspend release of radioactive effluents via this pathway.

Table 3.20-1 (continued)

RADIOACTIVE GASES EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Number of Channels Operable	Action
3. Radwaste Service Area Vent*		
a. Noble Gas Activity Monitor	1	With the monitor inoperable, effluent releases via this pathway may continue provided grab samples are taken at least once per 8 hours and these samples are analyzed in accordance with Table 4.22-1 within 24 hours.
b. Iodine Monitor	1	With the collection device inoperable, effluent releases via this pathway may continue provided continuous samples are taken and these samples are analyzed in accordance with Table 4.22-1 within 24 hours.
c. Particulate Monitor	1	With the collection device inoperable, effluent releases via this pathway may continue provided continuous samples are taken and these samples are analyzed in accordance with Table 4.22-1 within 24 hours.
d. System Effluent Flow Rate Device	1	With the flow device inoperable effluent releases via this pathway may continue provided the flow rate used is the maximum design flow rate.
e. Sampler Flow Rate Measurement Device	1	With the flow rate device inopera-able, effluent releases via this pathway may continue provided the flow rate is estimated and recorded at least once per 4 hours.

* The Radwaste Service Area Vent Monitoring System is not yet functional. This specification for this system will become effective when it is declared OPERABLE.

3.21 LIQUID EFFLUENTS

3.21.1 Concentration

The concentration of radioactive material released at any time beyond the site boundary shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} μCi/ml.

Applicability At all times

Action

With the concentration of radioactive material released from the site to unrestricted areas exceeding Specification 3.21.1, restore concentration within the specification limits as soon as practicable and provide notification to the Commission per the applicable sections of 10 CFR 20.403 and 10 CFR 20.405.

Bases

This Specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents from the site to areas beyond the site boundary will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II. This limitation provides additional assurance that the levels of radioactive materials in bodies of water outside the site will not result in exposures within: (1) the Section II. A Design Objectives of Appendix I, 10 CFR Part 50, to an individual, and (2) the limits of 10 CFR Part 20.106 (e) to the population. The concentration limit for dissolved or entrained noble gases is based upon the assumption that Xe-135 is the controlling radioiotope and its MPC in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

3.21.2 Dose

The dose or dose commitment to a member of the public from radioactive materials in liquid effluents released beyond the site boundary shall be limited:

a. During any calendar quarter to 1.5 mrem to the total body and to 5 mrem any organ; and

b. During any calendar year to 3 mrem to the total body and to 10 mrem to any organ.

Applicability At all times

Action

a. With the calculated dose or dose commitment from the release of radioactive materials in liquid effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days a Special Report. This Report will identify the cause(s) for exceeding the limit and define the corrective actions to be taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

Bases

This specification is provided to implement the requirements of Sections II.A, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents will be kept "as low as is reasonably achievable." The dose calculations in the ODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

3.21.3 Liquid Holdup Tanks

The quantity of radioactive material contained in each of the following tanks shall be limited to less than or equal to 10 Curies, excluding tritium and dissolved or entrained noble gases:

a. Regenerant Holdup Tanks

b. Outside Temporary Tanks

Applicability At all times

Action

With the quantity of radioactive material in any of the listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank and reduce the quantity to within the limit as soon as practicable.

Bases

Restricting the quantity of radioactive material contained in the specified outdoor tanks provides assurance that in the event of an uncontrolled release of the contents, the concentration at the nearest surface water supply in an unrestricted area would be less than the limits of 10 CFR 20, Appendix B, Table II, Column 2. There are two Regenerant Holdup Tanks. The limit applies to each tank individually.

3.22 GASEOUS EFFLUENTS

3.22.1 Dose Rate

The dose rate at and beyond the site boundary due to radioactive materials released in gaseous effluents from the site shall be limited to the following values:

a. The dose rate limit for noble gases shall be 500 mrem/yr to the total body and 3000 mrem/yr to the skin.

b. The dose rate limit for all radioiodines and for all radioactive materials in particulate form and radionuclides other than noble gases with half lives greater than 8 days shall be 1500 mrem/yr to any organ.

Applicability At all times

Action

With the dose rate(s) exceeding the above limits, decrease the release rate as soon as practicable to comply with the limit(s) given in Specification 3.22.1 and provide notification to the Commission per the applicable sections of 10 CFR 20.403 and 10 CFR 20.405.

Bases

This specification is provided to ensure that the dose rate at anytime at the site boundary (see Figure 3.22-1) from gaseous effluents will be within the annual dose limits of 10 CFR Part 20 for unrestricted areas. The annual dose limits are the doses associated with the concentration of 10 CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of an individual outside the restricted area, either within or outside the site boundary, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106 (b)). For individuals who may at times be within the site boundary, the occupancy of the individual will be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the restricted area boundary. The specified release rate limits restrict at all times the corresponding gamma and beta dose rates above background to an individual at or beyond the restricted area boundary to 500 mrem/yr to the total body or to 3000 mrem/yr to the skin. These release rate limits also restrict at all times the corresponding thyroid dose rate above background to an infant via the cow-milk-infant pathway to 1500 mrem/yr for the nearest cow to the plant.

3.22.2 Noble Gases

The air dose at and beyond the site boundary due to noble gases released in gaseous effluents shall be limited to the following:

a. During any calendar quarter, to 5 mrad for gamma radiation and 10 mrad for beta radiation.

b. During any calendar year, to 10 mrad for gamma radiation and 20 mrad for beta radiation.

Applicability At all times

Action

a. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days a Special Report. This Report will identify the cause(s) for exceeding the limit(s) and define the corrective actions to be taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

Bases

This specification is provided to implement the requirements of Sections II.B, III.A, and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents will be kept "as low as is reasonably achievable". The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conform with the guides of Appendix I to be shown by calculational procedures based on models and data such that the actual exposure of an individual through the appropriate pathways is unlikely to be substantially underestimated. The dose calculations established in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents will be consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Release of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision I, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors", Revision 1, July 1977. The ODCM equations provided for determining the air doses at or beyond the restricted area boundary (see Figure 3.22-1) will be based upon the historical average atmospheric conditions.

3.22.3 Iodine -131, Tritium and Radionuclides in Particulate Form

The dose or dose commitment to a member of the public from I-131, from tritium, and from radionuclides in particulate form with half-lives greater than eight days in gaseous effluents released at and beyond the site boundary shall be limited to the following:

a. During any calendar quarter to 7.5 mrem to any organ.

b. During any calendar year to 15 mrem to any organ.

Applicability At all times

Action

With the calculated dose or dose commitment from the release of I-131, tritium, and radionuclides in particulate form with half-lives greater then eight days in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days a Special Report. This Report will identify the cause(s) for exceeding the limit and define the corrective actions to be taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

Bases

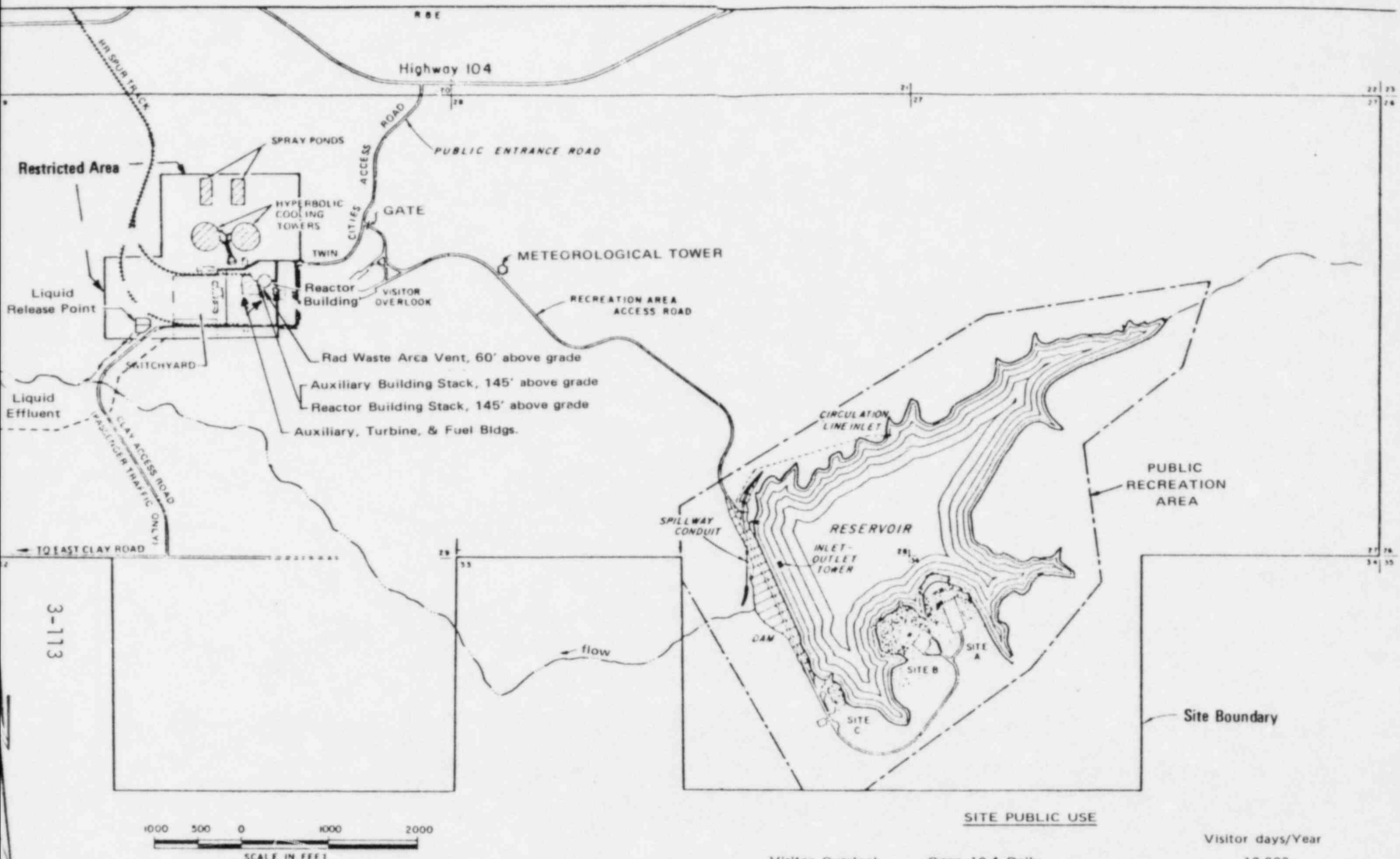
This specification is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section II.C of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." The ODCM calculational methods specified in the surveillance requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. For individuals who may at times be within the site boundry, the occupancy of the individual will be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the restricted area boundary. The ODCM calculational methods for calculating the doses due to the actual release rates of the subject materials are required to be consistent with the methodology provided in Regulatory Guide 1.109, "Calculating of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I", Revision I, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors", Revision 1, July 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions.

3.22.3 (Continued)

Bases (Continued)

The release rate specifications for radioiodines and particulates are dependent on the existing radionuclide pathways to man, beyond the site boundary. The pathways which were examined in the development of these calculations are: (1) individual inhalation of airborne radionuclides, (2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, (3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and (4) deposition on the ground with subsequent exposure of man.

FIGURE 3.22-1
GENERAL LAYOUT OF SITE



SITE PUBLIC USE

		Visitor days/Year
Visitor Overlook	Open 10-4, Daily	12,000
Recreation Area	Open 7 AM to Dark, Daily	180,000

3.23 GASEOUS RADWASTE TREATMENT

The gaseous radwaste treatment system and the ventilation exhaust treatment system shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected gaseous effluent air doses due to noble gas releases at and beyond the site boundary (see Figure 3.22-1), would exceed 0.2 mrad for gamma radiation and 0.4 mrad for beta radiation over 31 days. The ventilation exhaust treatment system shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected doses due to gaseous effluent releases from the site to areas at or beyond the site boundary would exceed 0.3 mrem to any organ over 31 days.

Applicability: When Gaseous Radwaste Treatment System and/or Ventilation Exhaust Treatment System are not being used.

Action

a. With gaseous waste being discharged without treatment and in excess of the above limits, prepare and submit to the Commission within 30 days, a Special Report which includes the following information:

   1. Explanation of why gaseous radwaste was being discharged without treatment, identification of the equipment or subsystems not OPERABLE and the reason for inoperability.

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status.

   3. Summary description of action(s) taken to prevent a recurrence.

Bases

The OPERABILITY of the gaseous radwaste treatment system and the ventilation exhaust treatment systems ensures that the systems will be available for use whenever gaseous effluents require treatment prior to release to the environment. The requirement that the appropriate portions of these systems be used when specified provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." The specification implements the requirements of 10 CFR Part 50.36 A, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and design objective Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the guide set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

3.24 GAS STORAGE TANKS

The quantity of radioactivity contained in each waste gas decay tank shall be limited to 135,000 curies of noble gases (considered as Xe-133).

Applicability At all times

Action

When the reactor coolant system activity reaches the limit of technical specification 3.1.4, sample the online waste gas decay tank daily to ensure that the limit of 135,000 curies equivalent Xe-133 is not exceeded .

Bases

Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the tanks contents, the resulting total body exposure to an individual at the nearest exclusion area boundary will not exceed 500 mrem. This is consistent with Standard Review Plan 15.7.1, "Waste Gas System Failure".

Potential atmospheric releases from a waste gas decay tank are evaluated assuming design coolant activities (see page 14D-25 Vol. VI FSAR). Based on primary coolant activity as shown in Table 14D-7, the decay tank is assumed to hold the activity associated with the off-gas from one reactor coolant system degassing with no credit taken for decay.

Calculation of the limiting decay tank activity based on the coolant activity limit of Technical Specification 3.1.4 yields a maximum decay tank inventory of 98,414 Ci (Ref. FSAR Table 14D-23) In order for the decay tank inventory to reach the limiting condition for operation, coolant activity would have to exceed the Technical Specification 3.1.4 limit on coolant activity and this would require a reactor shutdown, thus preventing a further increase in gaseous activity.

Therefore, it is conservative to require that the online waste gas decay tank be sampled daily upon reaching the coolant limiting activity value ($43/\bar{E}$) to insure the 135,000 curies equivalent Xe-133 is not exceeded. Once the coolant is below the limiting activity, there is no requirement to sample waste gas decay tanks except for discharging.

3.25 SOLID RADIOACTIVE WASTES

The solid radwaste systems shall be used in accordance with a PROCESS CONTROL PROGRAM to process wet radioactive wastes to meet shipping and burial requirements.

Applicability At all times.

Action

With the provisions of the PROCESS CONTROL PROGRAM not satisfied, suspend shipments of defectively processed or defectively packaged solid radioactive wastes from the site.

Bases

The OPERABILITY of the solid radwaste system ensures that the system will be available for use whenever radwastes require processing and packaging prior to being shipped offsite. This specification implements the requirements of 10 CFR 50.36a and General Design Criteria 60 of Appendix A to 10 CFR 50. The process parameters used in establishing the PROCESS CONTROL PROGRAM may include, but are not limited to waste type, waste pH, waste/solidification agent/catalyst ratios, waste oil content, waste principal chemical constituents, mixing and curing times.

3.26 RADIOLOGICAL ENVIRONMENTAL MONITORING

The radiological environmental monitoring program shall be conducted as specified in Table 3.26-1.

Applicability At all times

Action

a. With the radiological environmental monitoring program not being conducted as specified in Table 3.26-1, prepare and submit to the Commission, in the Annual Radiological Operating Report, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence. (Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, or to malfunction of automatic sampling equipment. If the latter, efforts shall be made to complete corrective action prior to the end of the next sampling period).

b. With the level of radioactivity in an environmental sampling medium exceeding the level of Table 3.26-2 when averaged over any calendar quarter, prepare and submit to the Commission within 30 days after the level of radioactivity has been determined, a Special Report which includes an evaluation of any release conditions, environmental factors or other aspects which caused the limits to be exceeded. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

c. With milk or fresh leafy vegetable samples unavailable from any of the sample locations required by Table 3.26-1, prepare and submit to the Commission within 30 days a Special Report which identifies the cause of the unavailability of samples and identifies locations for obtaining replacement samples. The locations from which samples were unavailable may then be deleted from Table 3.26-1 provided the locations from which the replacement samples were obtained are added to the environmental monitoring program as replacement locations, if available.

3.26 (Continued)

Bases (Continued)

The radiological monitoring program required by this specification provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures of individuals resulting from the station operation. This monitoring program thereby supplements the radiological effluent monitoring program by verifying that the measureable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and modeling of the environmental exposure pathways. The specified monitoring program is in effect at this time. Program changes may be initiated based on operational experience, and changes in regional population or agricultural practices. The sample locations have been listed in the ODCM to retain flexibility for making changes as needed.

With no drinking water intakes downstream of the plant, surface water and runoff water samples do not have to meet drinking water requirements and sample frequencies.

Limiting Conditions for Operation

TABLE 3.26-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples*	Sampling and Collection Frequency	Type and Frequency of Analysis
1. AIRBORNE			
A. Radioiodine and Particulates	8	Continuous operation of sampler collection as required by dust loading but at least once per week.	Radioiodine canister. Analyze at least once weekly for I-131. Particulate sampler. Analyze for Gross Beta radioactivity greater than or equal to 24 hours following filter change. Perform gamma isotopic analysis on each sample where gross beta activity is greater than 10 times the appropriate control samples for the same sample period. Perform gamma isotopic analysis on composite (by location) sample at least once per quarter.
2. DIRECT RADIATION	Greater than 40 locations with 2 dosimeters at each location.	At least once per quarter.	Gamma dose. At least once per quarter.

* Sample locations are shown in the ODCM.

Limiting Conditions for Operation

TABLE 3.26-1 (continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples*	Sampling and Collection Frequency	Type and Frequency of Analysis
3. WATERBORNE			
a. Surface	3	Grab sample collected monthly.	Gross Beta and I-131 analysis of each suspended and dissolved fraction. Tritium analysis at least once per quarter.
b. Runoff	1	Grab sample collected fortnightly.	Gross Beta and I-131 analysis of each suspended and dissolved fraction. Tritium analysis at least once per quarter, plus gamma isotopic analysis on dissolved and suspended fractions.
c. Mud and Silt	2	At least once semi-annually. One pint sample of the top 3" of material 2 ft. from shoreline.	Gross Beta on each sample.

* Sample locations are shown in the ODCM.

Limiting Conditions for Operation

TABLE 3.26-1 (continued)
RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples*	Sampling and Collection Frequency	Type and Frequency of Analysis
4. INGESTION			
a. Milk	4	At least once per fortnight when animals are on pasture; at least once per month at other times.	I-131 analysis of each sample.**
b. Fish	1	At least semi-annually. One sample of each of several species as shown in the ODCM.	Gross Beta minus K-40 analysis on edible portion of each sample.**
c. Food	4	At time of harvest. One sample of each of the several classes of food products as shown in the ODCM.	Gross Beta minus K-40 analysis on edible portion of each sample.**

*Sample locations are shown in the ODCM.
**Gamma Isotopic Analysis when Table 3.26-2 levels are exceeded.

Table 3.26-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

Reporting Levels

Analysis	Water (pCi/l	Airbore Particulate or Gases (PCi/m^3)	Fish (pCi/gm, dry)	Milk (pCi/l	Food Products (pCi/gm, dry)
H-3	2×10^4				
Mn-54	1×10^3				
Fe-59	4×10^2				
Co-58	1×10^3				
Co-60	3×10^2				
Zn-65	3×10^2				
Zr-Nb-95	4×10^2				
I-131	2	0.9		3	
Cs-134	30	10			
Cs-137	50	20			
Ba-La-140	2×10^2				
Gross beta	40	2	10		10

Table 3.26-3

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS [(a)]

Lower Limit of Detection (LLD) [(b)]

Analysis	Fish (pCi/gm, dry)	Milk (pCi/e)	Food Products (pCi/gm, dry)
Mn-54	0.03		
Fe-59	0.03		
Co-58	0.02		
Co-60	0.06		
Zn-65	0.06		
I-131		0.5	0.03
Cs-134	0.08	16	0.08
Cs-137	0.06	11	0.06
Ba-La-140		15	
Gross Beta	0.02		0.02

(a) This list does not mean that only these nuclides will be detected and reported. Other peaks that are measurable and identifiable, together with the above nuclides, will be identified and reported.

(b) LLD is defined in the ODCM.

3.27 LAND USE CENSUS

A land use census shall be conducted annually and shall identify the location of the nearest milk animal, the nearest residence and the nearest garden* of greater than 500 square feet producing fresh leafy vegetables in each of the 16 meteorological sectors within a distance of five miles.

Applicability At all times

Action

a. With a land use census identifying a location(s) which yields a calculated dose or dose commitment greater than the values currently being calculated in Specification 4.22.3, identify the new locations in the next Semiannual Radioactive Effluent Release Report.

b. With a land use census identifying a location(s) that yields a calculated dose or dose commitment (via the same exposure pathway) 20% greater than at a location from which samples are currently being obtained in accordance with Specification 3.26, add the new location(s) to the radiological environmental monitoring program within 30 days. The sampling location(s), excluding the control station location, having the lowest calculated dose or dose commitment(s) (via the same exposure pathway) may be deleted from this monitoring program after (October 31) of the year in which this land use census was conducted. Identify the new location(s) in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

*Broad leaf vegetation sampling may be performed at the site boundary in the direction sector with the highest X/Q in lieu of the garden census.

3.27 (Continued)

Bases

This specification is provided to ensure that changes in the use of areas at and beyond the site boundary are identified and that modifications to the monitoring program are made if required by the results of this census. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. Restricting the census to gardens of greater than 500 square feet provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity (26 kg/yr) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were used: (1) that 20% of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage); and (2) a vegetation yield of 2 kg/square meter.

3.28 EXPLOSIVE GAS MIXTURE

The concentration of oxygen in the waste gas holdup system shall be limited to $\leq$4% by volume.

Applicability

At all times.

Action

With the concentration of oxygen in the waste gas holdup system $>$4% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to $\leq$4% within 48 hours.

Bases

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the waste gas treatment system is maintained below the flammability limits of hydrogen and oxygen. Maintaining the concentration of oxygen below the flammability limit provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

3.29 FUEL CYCLE DOSE

The annual dose or dose commitment to a member of the public due to releases of radioactivity and radiation from uranium fuel cycle sources is limited to ≤ 25 mrem to the total body or any organ (except the thyroid, which is limited to ≤75 mrem).

Applicability At all times.

Action

With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specifications 3.21.2a, 3.21.2b, 3.22.1a, 3.22.1b, 3.22.2a, 3.22.2b, 3.22.3a, or 3.22.3b, calculations should be made to determine whether the above limits of Specification 3.29 have been exceeded. If such is the case, prepare and submit to the Commission within 30 days, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR Part 20.405c, shall include an analysis that estimates the radiation exposure (dose) to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceed the above limits, and if the release condition resulting in violation of 40 CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provision of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

Bases

This specification is provided to meet the dose limitations of 40 CFR 190. The specification requires the preparation and submittal of a Special Report whenever the calculated doses from plant radioactive effluents exceed twice the design objective doses of Appendix I. For the Rancho Seco site it is highly unlikely that the resultant dose to a MEMBER OF THE PUBLIC will exceed the dose limits of 40 CFR 190 if the plant remains within the reporting requirement level. The Special Report will describe a course of action which should result in the limitation of the annual dose to a MEMBER OF THE PUBLIC to within the 40 CFR 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the MEMBER OF THE PUBLIC from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 5 miles must be considered. If the dose to any MEMBER OF

3.29 (Continued)

Bases (Continued)

THE PUBLIC is estimated to exceed the requirements of 40 CFR 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR 190 have not already been corrected), in accordance with the provisions of 40 CFR 190.11 is considered to be a timely request and fulfills the requirements of 40 CFR 190 until NRC staff action is completed. An individual is not considered a member of the public during any period in which he/she is engaged in carrying out any operation which is part of the nuclear fuel cycle.

3.30 INTERLABORATORY COMPARISON PROGRAM

The contractor performing the analysis of radiological environmental program samples for radioactive materials shall participate in an Inter-Laboratory Comparison Program approved by the Commission.

Applicability At all times

Action

With analyses not being performed as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report.

Bases

The requirement for participation in an Inter-Laboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental samples are performed as part of the quality assurance program for environmental monitoring in order to demonstrate that the results are reasonably valid.

4. SURVEILLANCE STANDARDS

4.1 OPERATIONAL SAFETY REVIEW

Applicability

Applies to items directly related to safety limits and limiting conditions for operation during power operation. During cold shutdown, systems and components required to maintain safe shutdown will be tested.

Objective

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

Specification

4.1.1 The minimum frequency and type of surveillance required for reactor protection system and safety feature protection system instrumentation when the reactor is critical shall be as stated in Table 4.1-1.

4.1.2 Equipment and sampling test shall be performed as detailed in Tables 4.1-2 and 4.1-3.

4.1.3 A power distribution map shall be made to verify the expected power distribution at periodic intervals on approximately every 10 effective full power days using the incore instrumentation detector system.

Bases

Check

Failures such as blown instrument fuses, defective indicators, faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures are, in many cases, revealed by alarm or annunciator action. Comparison of output and/or state of independent channels measuring the same variable supplements this type of built-in surveillance. Based on experience in operation of both conventional and nuclear systems, when the unit is in operation, the minimum checking frequency stated is deemed adequate for reactor system instrumentation.

4.1 (Continued)

Calibration

Calibration shall be performed to assure the presentation and acquisition of accurate information. The nuclear flux (power range) channels amplifiers shall be calibrated (during steady state operating conditions) against a heat balance standard when the indicated neutron power and core thermal power differ by more than two percent. During non-steady state operation, the nuclear flux channels amplifiers shall be calibrated daily to compensate for instrumentation drift and changing rod patterns and core physics parameters.

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

Substantial calibration shifts within a channel (essentially a channel failure) will be revealed during routine checking and testing procedures. Thus, minimum calibration frequencies set forth are considered acceptable.

Testing

The frequency of on-line testing of reactor protective channels as shown in Table 4.1-1 will assure the required level of performance.

The equipment testing and system sampling frequencies specified in Table 4.1-2 and Table 4.1-3 are considered adequate to maintain the equipment and systems in a safe operational status.[1]

Power Distribution Mapping

The incore instrumentation detector system will provide a means of assuring that axial and radial power peaks and the peak locations are being controlled by the provisions of the Technical Specifications within the limits employed in the safety analysis.

REFERENCES

(1) FSAR paragraph 1.4.12

Table 4.1-1

INSTRUMENT SURVEILLANCE REQUIREMENTS

Channel Description	Check	Test	Calibrate	Remarks
Reactor Protective System				
1. Source range channel	S (1)	P	NA	(1) When in service.
2. Intermediate range channel	S	P	NA	
3. Power range amplifier	D (1)	NA	(2)	(1) Heat balance check daily. (2) Heat balance calibration whenever indicated neutron power and core thermal power differ by more than 2% and daily during non-steady-state operation.
4. Power range channel	S	M	M (1,2)	(1) Using incore instrumentation for split detector calibration. (2) Imbalance, upper and lower chambers at equilibrium xenon after each startup if not done the previous week.
5. High reactor coolant pressure channel	S	M	R	
6. Low reactor coolant pressure channel	S	M	R	
7. Reactor coolant temperature channel	S	M	R	

S = Each shift
D = Daily
W = Weekly
M = Monthly
Q = Quarterly
P = Prior to each startup if not done previous week
R = Once during the refueling interval

Table 4.1-1 (Continued)

INSTRUMENT SURVEILLANCE REQUIREMENTS

Channel Description	Check	Test	Calibrate	Remarks
8. Reactor coolant pressure/ temperature comparator	S	M	R	
9. Power/imbalance/flow comparator	S	M	R	
10. Pump/flux comparator	S	M	R	
11. High Reactor Building Pressure Channels	D	M	R	
12. Protection channel coincidence logic	NA	M	NA	
13a. CRD Trip Breaker				
(1) RPS Undervoltage trip	NA	M	NA	
(2) Turbine/Generator, Loss of Feedwater Trip	NA	M	NA	
13b. Turbine Generator Trip Functional Test	NA	SY (1)	R	(1) Test at next cold shutdown.
Safety Features System				
14. Emergency core cooling injection, emergency building cooling and building isolation analog channels				
a. Reactor coolant pressure channel	S	M	R	
b. Reactor Building 4 psig channel	S	M	R	

S = Each shift
M = Monthly
P = Prior to each startup if not done previous week
D = Daily
Q = Quarterly
R = Once during the refueling interval
W = Weekly

Table 4.1-1 (Continued)

INSTRUMENT SURVEILLANCE REQUIREMENTS

Channel Description	Check	Test	Calibrate	Remarks
15. Reactor Building spray system analog channels				
a. Spray pump 30 psig channels	NA	M	R	
b. Spray valve 30 psig channels	NA	M	R	
16. High pressure injection, emergency building cooling and building isolation digital logic channels	NA	M	NA	
17. Low pressure injection digital logic channels	NA	M	NA	
18. Reactor Building spray pumps digital logic channels	NA	M	NA	
19. Reactor Building spray valves digital logic channels	NA	M	NA	

S = Each shift
M = Monthly
P = Prior to each startup if not done previous week

D = Daily
Q = Quarterly
R = Once during the refueling interval

W = Weekly

Table 4.1-1 (Continued)

INSTRUMENT SURVEILLANCE REQUIREMENTS

	Channel Description	Check	Test	Calibrate	Remarks
20.	High pressure injection, Reactor Building isolation, and Reactor Building emergency cooling Channel A manual trip	NA	M	NA	
21.	High pressure injection Reactor Building isolation, and Reactor Building emergency cooling Channel B manual trip	NA	M	NA	
22.	Low pressure injection Channel A manual trip	NA	R	NA	
23.	Low pressure injection Channel B manual trip	NA	R	NA	
24.	Reactor Building spray pump Channel A manual trip	NA	R	NA	
25.	Reactor Building spray pump Channel B manual trip	NA	R	NA	
26.	Reactor Building spray valves Channel A manual trip	NA	R	NA	
27.	Reactor Building spray valves Channel B manual trip	NA	R	NA	

S = Each shift
D = Daily
W = Weekly
M = Monthly
Q = Quarterly
P = Prior to each startup if not done previous week
R = Once during the refueling interval

Table 4.1-1 (Continued)

INSTRUMENT SURVEILLANCE REQUIREMENTS

Channel Description	Check	Test	Calibrate	Remarks
Process Instrumentation				
28. Core flooding tanks				
a. Pressure channels	D	NA	R	
b. Level channels	D	NA	R	
29. Pressurizer level channels	D	NA	R	
30. Pressurizer temperature channels	S	NA	R	
31. Make-up tank level channels	D	NA	R	
32. High pressure injection flow channels	NA	NA	R	
33. Low pressure injection flow channels	NA	NA	R	
34. Borated water storage tank level indicator	W	Q	R	

S = Each shift
M = Monthly
P = Prior to each startup if not done previous week
D = Daily
Q = Quarterly
R = Once during the refueling interval
W = Weekly

Table 4.1-1 (Continued)

INSTRUMENT SURVEILLANCE REQUIREMENTS

Channel Description	Check	Test	Calibrate	Remarks
35. Spray additive tank				
a. Level channel	W	NA	R	
36. Concentrated boric acid storage tank				
a. Level channel	W	NA	R	
b. Temperature channel	M	NA	R	
37. Steam generator water level	W	NA	R	
38. Control rod absolute position	S (1)	NA	R (2)	(1) Check with relative position indicator. (2) Calibrate rod misalignment channel.
39. Control rod relative position	S (1)	NA	R (2)	(1) Check with absolute position indicator. (2) Calibrate rod misalignment channel.
40. Reactor Building temperature	NA	NA	R	
41. Reactor Building emergency sump level alarm	NA	NA	R	

S = Each shift
D = Daily
W = Weekly
M = Monthly
Q = Quarterly
P = Prior to each startup if not done previous week
R = Once during the refueling interval

Table 4.1-1 (Continued)

INSTRUMENT SURVEILLANCE REQUIREMENTS

Channel Description	Check	Test	Calibrate	Remarks
42. Reactor Building drain accumulation tank level	NA	NA	R	
43. Incore neutron detectors	M (1)	NA	NA	(1) Check functioning, including functioning of computer readout and/or recorder readout.
44. Process and area radiation monitoring system	W	M	Q	
45. Emergency plant radiation instruments	M (1)	NA	R	(1) Battery Check.
46. Environmental air monitors	M (1)	NA	R	(1) Check functioning.
47. Strong motion accelerometer	Q (1)	NA	R	(1) Battery Check.
48. Auxiliary Feedwater Start Circuit				
a. Phase Imbalance/Underpower RCP	S	M	R	
b. Low Main Feedwater Pressure	NA	NA	R	
49. Pressurizer Water Level	M	NA	R	
50. Auxiliary Feedwater Flow Rate	M	NA	R	
51. Reactor Coolant System Subcooling Margin Monitor	M	NA	R	

S = Each shift
D = Daily
W = Weekly
M = Monthly
Q = Quarterly
P = Prior to each startup if not done previous week
R = Once during the refueling interval

Table 4.1-1 (Continued)

INSTRUMENT SURVEILLANCE REQUIREMENTS

Channel Description	Check	Test	Calibrate	Remarks
52. EMOV Power Positive Indicator (Primary Detector)	M	NA	R	
53. EMOV Position Indicator (Backup Detector) T/C or Acoustic	M	NA	R	
54. EMOV Block Valve Position Indicator	M	NA	R	
55. Safety Valve Position Indicator (Primary Detector) T/C	M	NA	R	
56. Safety Valve Position Indicator (Primary Detector) Acoustic	M	NA	R	

S = Each shift
D = Daily
W = Weekly
M = Monthly
Q = Quarterly
P = Prior to each startup if not done previous week
R = Once during the refueling interval

TABLE 4.1-2

MINIMUM TEST FREQUENCY

Item	Test	Frequency
1. Control rods	Rod drop times of all full length rods	Each refueling shutdown
2. Control rod movement	Movement of each rod	Every two (2) weeks
3. Pressurizer code safety valves	Setpoint	1 each refueling interval
4. Main steam safety valves	Setpoint	2 per steam generator each refueling interval
5. Refueling system interlocks	Functional	Each refueling interval prior to handling fuel
6. Turbine steam stop valves	Movement of each valve	Monthly
7. Reactor coolant system	Leakage	Calculated inventory weekly. Leakage check daily
8. Charcoal & high efficiency filters	Charcoal & HEPA filter for iodine & particulate removal efficiencies. DOP test on HEPA filters. Freon test on charcoal filter units.	Each refueling interval and at any time work on filters could alter their integrity
9. Fire pumps and power supplies	Functional	Monthly
10. Reactor Building isolation trip	Functional	Each refueling interval
11. Spent fuel cooling system	Functional	Each refueling interval prior to fuel handling

. . . . Continued

TABLE 4.1-2 (Cont'd.)

MINIMUM TEST FREQUENCY

Item	Test	Frequency
12. Turbine overspeed trip	Calibration	Each refueling interval
13. Internals vent valves	Manual actuation$^{(1)}$, remote visual inspection$^{(2)}$, verify valve not stuck open	Each refueling interval

1. Verifying through manual actuation that the valve is fully open with a force of 400 lbs. (applied vertically upward).

2. Check visually accessible surfaces to evaluate observed surface irregularities.

TABLE 4.1-3
MINIMUM SAMPLING FREQUENCY

Item		Check	Frequency
1. Reactor coolant	a.	Radio-chemical analysis(1)	M
		$\bar{E}$ determination(4) (3)	Semiannually
	b.	Gross activity(1) (3)	3/week
	c.	Tritium radioactivity(3)	M
	d.	Chemistry (Cl and O_2)(3)	3/week
	e.	Boron concentration	2/week
	f.	Fluoride(3)	M
2. Borated water storage tank water sample		Boron concentration	After each makeup but at least monthly
3. Core flooding tank water sample		Boron concentration(3)	After each makeup but at least monthly
4. Spent fuel storage water sample		Boron concentration	After each makeup but at least monthly
5. Secondary coolant	a.	Gross activity(2)(3)	Weekly
	b.	Iodine analysis(3)	Weekly
6. Concentrated boric acid tank		Boron concentration	After each makeup but at least twice weekly

. . . . . . Continued

TABLE 4.1-3 (Cont'd.)
MINIMUM SAMPLING FREQUENCY

Item	Check	Frequency
7. Spray additive tank	NaOH Concentration(3)	After each makeup but at least quarterly.
8. Blowdown from cooling towers	Gross activity (3)	M

(1) When radioactivity level is greater than 20 percent of the limits of Technical Specification 3.1.4, the sampling frequency shall be increased to a minimum of once each day.

(2) When gross activity increases to $5x10^{-8}$ μCi/cc above normal, an iodine analysis will be made and performed thereafter when the gross activity increases by a factor of 10 percent.

(3) Not performed during cold shutdown.

(4) $\bar{E}$ determination will be started when gross beta-gamma activity analysis indicates greater than 10 μCi/ml and will be redetermined each 10 μCi/ml increase in gross beta-gamma activity analysis. A radio-chemical analysis for this purpose shall consist of a quantitative measurement of 95% of radionuclides in reactor coolant with half lives of $>$30 minutes.

4.2 REACTOR COOLANT SYSTEM SURVEILLANCE

Applicability

Applies to the reactor vessel, the reactor coolant system and its components.

Objective

To establish examinations whereby the reactor coolant system and component integrity is monitored.

Specification

4.2.1 The program for irradiation surveillance of the reactor vessel materials to monitor changes in the mechanical and impact properties shall be performed as described in paragraph 4.4.5 of the FSAR. Removal of specimens from capsules within the reactor shall be as scheduled in Table 4.2-1, SMUD shall be responsible for testing the specimens and submitting a report of test results in accordance with 10 CFR 50, Appendix H.

4.2.2 An inservice inspection shall be made conforming as closely as design permits to the rules of the ASME Boiler and Pressure Vessel Code Section XI, Rules for Inservice Inspection of Nuclear Reactor Coolant Systems with revisions approved as of June 1973, Tables IS-261, IS-251, and Section IS-240 of this Code will be used as a guide for determining the examination frequencies and the applicable specific areas to be examined. The inspection interval will be ten years. As part of the inservice inspection, hydrostatic tests will be performed as prescribed under Section IS-500 of this Code.

4.2.3 A preoperational examination will be made to include all the items that would normally be completed throughout the inspection interval. This survey will establish initial system integrity and provide a baseline for future testing.

4.2.4 Each reactor coolant pump motor flywheel will be inspected volumetrically during the ten-year inspection interval. One hundred percent of the flywheel will be examined. All flywheels received a one hundred percent ultrasonic examination prior to installation of the motor.

Because the reactor coolant system was not designed to meet the requirements of Section XI of the ASME Boiler and Pressure Vessel Code, complete compliance is not feasible or practical. However, access for inservice inspection has been considered and design modifications made where practical.

Therefore, where possible, Section XI of this Code will be utilized in the conduct of this program. Table 4.2.2 itemizes those areas where complete compliance with the code is not possible because of specific design and construction details.

4.2.5 If as a result of any of these inspections, defects are found to develop, further examinations will be made as needed to determine the exact condition. Following evaluation of this evidence, a decision will be made to the effect upon plant safety and the requirements for repairs.

4.2.6 Records of each inspection shall be kept to permit evaluation and future comparison.

4.2.7 Periodic consideration will be given to incorporation of new or improved inspection techniques into the surveillance program.

4.2.8 A report or application for license amendment shall be submitted to the NRC within 90 days after the occurrence of any of the following:

1. Failure of Davis-Besse Unit No. 1 to achieve commercial operation at 100% power by January 1, 1978

2. Beginning one year after attainment of commercial operation 100% power, any time that Davis-Besse Unit No. 1 fails to maintain a cumulative reactor utilization factor of greater than 65%.

The report shall provide justification for continued operation of Rancho Seco with the reactor vessel surveillance program conducted at Davis-Besse Unit No. 1 or the application for license amendment shall propose an alternative program for conduct of the Rancho Seco reactor vessel surveillance program.

Bases

Irradiation surveillance provides the capability of determining the radiation-induced changes in the mechanical and impact properties in the region of the reactor vessel surrounding the core. Test specimens of base metal, deposited weld metal and the heat-affected zone are installed in capsule assemblies placed inside the vessel. In accordance with the schedules of Table 4.2-1 specimens will be removed; and a series of drop weight tests, Charpy impact tests and tension tests will be conducted. Threshold neutron flux detectors and maximum temperature detectors will be installed with the specimens. Changes in nil-ductility transition temperature will be determined and appropriate alteration to plant operating parameters will be made.

Preoperational and inservice inspections emphasize areas of highest stress concentration and probability of failure. The area predominantly selected for these examinations are welds and the adjacent metal. Examination of the welds is often by a volumetric (ultrasonic or radiography) method which, when performed, examines surrounding base metal and the weld heat-affected zone. Both testing methods will use present state-of-the-art equipment operated by highly trained personnel qualified within the requirements of the applicable codes.

4.2 (Continued)

Bases (Continued)

To assure the availability of adequate surveillance data for the Rancho Seco No. 1 reactor vessel, a program has been developed to monitor the irradiation of the surveillance specimen capsules at the Davis-Besse No. 1 reactor, and compare this to the irradiation of the Rancho Seco No. 1 reactor vessel. Fluence estimates which are conservative in the appropriate direction are used for this comparison. The frequency of monitoring varies depending on the known neutron fluence lead factor between the capsules and the reactor vessel. This provides ample time for anticipating problems and initiating corrective action should operation of the host reactor be seriously delayed. For the purpose of Technical Specification 4.2.8, the definition of Regulatory Guide 1.16, Revision 4, (August 1975) applies for the term "commercial operation". Cumulative reactor utilization factor is defined as: [(Cumulative thermal megawatt hours since attainment of commercial operation at 100% power) x 100] ÷ [(licensed thermal power) x (cumulative hours since attainment of commercial operation at 100% power)].

TABLE 4.2-1
RANCHO SECO CAPSULE ASSEMBLY WITHDRAWAL SCHEDULE AT DAVIS-BESSE 1

CAPSULE	INSERTION/WITHDRAWAL
RSI-B	Withdraw at end of first cycle
RSI-E	Insert at end of first cycle, withdraw at end of tenth cycle
RSI-D	Withdraw at end of second cycle
RSI-A	Insert at end of second cycle, withdraw at end of seventh cycle
RSI-C	Withdraw at end of twelfth cycle
RSI-F	Withdraw at end of ninth cycle

TABLE 4.2-2
INSERVICE INSPECTION SCHEDULE

IS-261 Item	Component	Exception
1.2	Circumferential weld in reactor vessel bottom head	Weld may not be accessible because of radiation levels
5.1	Pump casing welds	Welds cannot be meaningfully volumetrically inspected
5.2	Pump casings	Will be inspected when a pump is disassembled for maintenance
5.6	Integrally welded supports	Lug welds on pump casings cannot be meaningfully volumetrically inspected

4.3 TESTING FOLLOWING OPENING OF SYSTEM

Applicability

Applies to test requirements for reactor coolant system integrity.

Objective

To assure reactor coolant system integrity prior to return to criticality following normal opening, modification, or repair.

Specification

4.3.1 When reactor coolant system repairs or modifications have been made, these repairs or modifications shall be inspected and tested to meet all applicable code requirements prior to the reactor being made critical.

4.3.2 Following any opening of the reactor coolant system, it shall be leak tested at not less than 2,255 psig prior to the reactor being made critical.

4.3.3 The limitations of Specification 3.1.2 shall apply.

Bases

Repairs or modifications made to the reactor coolant system are inspectable and testable under applicable codes, such as B31.7 and ASME Boiler and Pressure Vessel Code, Section XI, IS-400.

For normal opening, the integrity of the reactor coolant system, in terms of strength, is unchanged. If the system does not leak at 2,255 psig (operating pressure +100 psi; ±50 psi is normal pressure fluctuation), it will be leak tight during normal operation. (1)

REFERENCES

(1) FSAR, section 4

4.4 REACTOR BUILDING

4.4.1 Containment Leakage Tests

Applicability

Applies to containment leakage.

Objective

To verify that leakage from the Reactor Building is maintained within allowable limits.

Specification

4.4.1.1 Integrated Leakage Rate Tests

4.4.1.1.1 Calculated Peak Pressure Leakage Rate

The maximum allowable integrated leakage rate, L_a, from the Reactor Building at the 52 psig calculated peak containment internal pressure, Pp, shall not exceed 0.10 weight percent of the building atmosphere at that pressure per 24 hours.

4.4.1.1.2 Testing at Reduced Pressure

The periodic integrated leak rate test may be performed at a test pressure, Pt, of not less than 26 psig provided the resultant leakage rate, Lt, does not exceed a pre-established fraction of L_a determined as follows:

A. Prior to reactor operation the initial value of the integrated leakage rate of the Reactor Building shall be measured at Pp and at the reduced pressure (pt) to be used in the periodic integrated leakage rate tests. The leakage rates thus measured shall be identified as L_{pm} and L_{tm}, respectively.

B. L_t shall not exceed $L_a \left[\frac{L_{tm}}{L_{pm}}\right]$ for values of $\frac{L_{tm}}{L_{pm}}$ below 0.7.

C. L_t shall not exceed $L_a \sqrt{\frac{P_t}{P_p}}$ for values of $\frac{L_{tm}}{L_{pm}}$ above 0.7.

4.4.1.1.2 (Continued)

D. If $\frac{L_{tm}}{L_{pm}}$ is less than 0.3, the initial integrated test results shall be subject to review by the NRC to establish an acceptable value of L_t.

4.4.1.1.3 Conduct of Tests

A. The test duration shall be at least 24 hours unless experience from at least two prior tests on similar vessels provides evidence of the adequacy of a shorter test duration.

B. 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 the validity of measurements.

C. Closure of containment isolation valves for the purpose of the test shall be accomplished by the means provided for normal operation of the valves without preliminary exercises or adjustment.

4.4.1.1.4 Frequency of Test

After the initial preoperational leakage rate test, two integrated leakage rate tests shall be performed at approximately equal intervals between each major shutdown for inservice inspection to be performed at 10 year intervals. In addition, an integrated test shall be performed at each 10 year interval, coinciding with the inservice inspection shutdown. The test shall coincide with a shutdown for major fuel reloading.

4.4.1.1.5 Conditions for Return to Criticality

A. If L_t is less than 75% of the value permitted in 4.4.1.1.2, local leakage rate testing need not be completed prior to return to criticality following a periodic integrated leakage rate test.

B. If L_t is between 75% and 100% of the value permitted in 4.4.1.1.2 the return to criticality will be permitted conditioned upon demonstration that local leakage rate measured at full design pressure, account for all leakage above 75% of L_t. If this cannot be demonstrated within 30 days of returning to criticality, the reactor shall be shutdown.

4.4.1.1.6 Corrective Action and Retest

If repairs are necessary to meet the criteria of 4.4.1.1.1 or 4.4.1.1.2, the integrated leak rate test need not be repeated provided local leakage rate measurements are made before and after repair to demonstrate that the leakage rate reduction achieved by repairs reduces the overall measured integrated leak rate to an acceptable value.

4.4.1.1.7 Report of Test Results

Each integrated leak rate test will be the subject of a summary technical report which will include a description of test methods used and a summary of local leak detection tests. Sufficient data and analysis shall be included to show that a stabilized leak rate was attained and to identify all significant required correction factors such as those associated with humidity and barometric pressure, and all significant errors such as those associated with instrumentation sensitivities and data scatter.

4.4.1.2 Local Leakage Rate Tests

4.4.1.2.1 Scope of Testing

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

(1) Personnel hatch
(2) Emergency hatch
(3) Equipment hatch seals
(4) Fuel transfer tube seals
(5) Fuel transfer tube shroud bellows
(6) Reactor Building normal sump drain line
(7) Reactor coolant pump seal water outlet line
(8) Reactor coolant pump seal inlet line
(9) Reactor Building equalizing line
(10) Decay Heat removal inlet lines
(11) Reactor Building spray inlet lines
(12) High pressure injection lines
(13) Electrical penetrations
(14) Reactor Building purge inlet line
(15) Reactor Building purge outlet line
(16) Reactor Building atmosphere sample lines
(17) Letdown to purification demineralizer line
(18) Pressurizer relief tank gas sample line
(19) Reactor coolant system vent header
(20) Pressurizer relief tank nitrogen supply line
(21) Pressurizer sample line
(22) Reactor coolant drain tank header

4.4.1.2.2 Conduct of Tests

(a) With the exception of the personnel and emergency hatches' door seals, all local leak rate tests shall be performed at a pressure of not less than 52 psig. See Paragraph 4.4.1.2.5c for personnel and emergency hatch door seal test pressure.

(b) Acceptable methods of testing are halogen gas detection, soap bubbles, pressure decay, hydrostatic flow or equivalent.

4.4.1.2.3 Acceptance Criteria

The total leakage from all penetrations and isolation valves shall not exceed 0.06 percent of the Reactor Building atmosphere per 24 hours.

4.4.1.2.4 Corrective Action and Retest

(a) If at any time it is determined that the criterion of 4.4.1.2.3 above is exceeded, repairs shall be initiated immediately.

(b) If conformance to the criterion of 4.4.1.2.3 is not demonstrated within 48 hours following detection of excessive local leakage, the reactor shall be shut down and depressurized until repairs are effected and the local leakage meets the acceptance criterion as demonstrated by retest.

4.4.1.2.5 Test Frequency

Local leak detection test shall be performed at a frequency of at least each refueling interval except that:

(a) The equipment hatch and fuel transfer tube seals shall be additionally tested after each opening.

(b) The personnel and emergency hatches shall be tested between the inner and outer doors at a pressure not less that 52 psig semi-annually.

(c) *The personnel and emergancy hatches' inner and outer door O-ring seals shall be tested within 72 hours after each opening when containment integrity is required in Specification 3.6.1. Test pressure for the personnel and emergency hatches' O-ring seals shall be 10 psig.

*Exemption to Appendix J. of 10 CFR 50.

4.4.1.2.5(c) (Continued)

(1) The leak rate (L_t) established at the reduced pressure of 10 psig shall be extrapolated to the leak rate (L_a) that will occur at the calculated peak containment pressure of 52 psig using the following formula:

$$L_a = 5.2\ L_t$$

(2) The extrapolated leak rate (L_a) will be added to the local leak rates established for the other components and the total must meet the criterion of 4.4.1.2.3.

4.4.1.3 Isolation Valve Functional Tests

Every three months, remotely operated Reactor Building isolation valves shall be stroked to the position required to fulfill their safety function unless such operation is not practical during plant operation. The latter valves shall be tested during the next shutdown of 72 hours duration.

4.4.1.4 Annual Inspection

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

4.4.1.5 Reactor Building Modifications

Any major modification or replacement of components affecting the Reactor Building integrity shall be followed by either an integrated leak rate test or a local leak test, as appropriate, and shall meet the acceptance criteria of 4.4.1.1.5 and 4.4.1.2.3 respectively.

Bases

The Reactor Building is designed for an internal pressure of 59 psig and a steam-air mixture temperature of 286 F. Prior to initial operation, the containment will be strength tested at 115 percent of design pressure. The containment will also be leak tested prior to initial operation at P_p and P_t (52 psig and 26 psig, respectively). These tests will verify that the leakage rate from Reactor Building pressurization satisfies the relationships given in the specification . (1)(2)

4.4.1 (Continued)

Bases (Continued)

The performance of a periodic integrated leakage rate test during plant life provides a current assessment of potential leakage from the containment in case of an accident that would pressurize the interior of the containment. In order to provide a realistic appraisal of the integrity of the containment under accident conditions, this periodic test is to be performed without preliminary leak detection surveys or leak repairs, and containment isolation 26 psig for the periodic integrated leakage rate test is sufficiently high to provide an accurate measurement of the leakage rate and it duplicates the pre-operational leakage rate test at 26 psig. The specification provides a relationship for relating the measured leakage of air at 26 psig to the potential leakage at 52 psig. The minimum of 24 hours was specified for the integrated leakage rate test to help stabilize conditions and thus improve accuracy and to better evaluate data scatter. The frequency of the periodic integrated leakage rate test is keyed to the refueling schedule for the reactor, because these tests can best be performed during refueling shutdowns.

The specified frequency of periodic integrated leakage rate tests is based on three major considerations. First is the low probability of leaks in the liner, because of conformance of the complete containment to a 0.10 percent leakage rate at 52 psig during pre-operational testing and the absence of any significant stresses in the liner during reactor operation. Second is the more frequent testing, at 52 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 0.06 percent of leakage that is specificed as acceptable from penetrations and isolation valves. Third is the tendon stress surveillance program which provides assurance that an imporatant part of the structural integrity of the containment is maintained.

More frequent testing of various penetrations is specified as these locations are more susceptible to leakage that the Reactor Building liner due to the penetrations with resilient sealing materials, penetrations that vent directly to the Reactor Building atmosphere, and penetrations that connect to the reactor coolant system pressure boundary. The basis for specification of a total leakage rate of (0.075 percent) from penetrations and isolation valves should be from those sources, in order to provide assurance that the integrated leakage rate would remain within the specified limits during the intervals between integrated leakage rate tests. Valve operability tests are specified to assure proper closure or opening of the Reactor Building isolation valves to provide for isolation of functioning of safety features systems. Valves will be stroked to the position required to fulfill their safety function unless it is established that such testing is not practical during operations.

4.4.1 (Continued)

Bases (Continued)

The airlock seals are tested at 10 psig because that is the manufacturer's recommended pressure for reverse flow through the seals. The extrapolation formula is derived assuming laminar, incompressible flow and provides conservative leak rates.

This specification complies with the Appendix J to 10 CFR 50 as published in the Federal Register on February 23, 1973, with the exemptions to Appendix J granted July 13, 1977

REFERENCES

(1) FSAR, Paragraph 5.2.1.1.1

(2) FSAR, Section 14

4.4.2 Structural Integrity

Applicability

Applies to the structural integrity of the Reactor Building.

Objective

To define the inservice surveillance program for the Reactor Building.

Specification

4.4.2.1 Tendon Surveillance

An inspection as described below for lift-off measurements, strand surveillance and anchorage surveillance shall be performed 1, 2, and 3 years after the initial containment integrity test and every 5 years thereafter.

4.4.2.2 Lift-Off Measurements

Lift-Off measurements of the prestress force shall be made on the following:

A. Six dome tendons, three normal and three modified with one of each in each 60^{0} group.

B. Six vertical tendons, three normal and three modified.

C. Six hoop tendons, three normal and three modified; modified hoop tendons are provided with shim stock at each anchorage to allow detensioning.

The lift-off readings shall not be less that the values predicted for the particular time the inspection is made. These predicted values shall be based upon the final jacking forces corrected for recorded seating losses and calculated losses due to concrete creep and shrinkage and prestressing steel relaxation.

After the intitial lift-off readings have been taken on the modified hoop tendons they shall be jacked to a force of .8f's and then compeletely detensioned to inspect for broken or damaged strands. Strand continuity will be verified during retensioning by comparison of predicted and observed elongations and forces.

All other surveillance tendons will be jacked only to observe and record lift-off.

4.4.2.3 Strand Surveillance

At each inspection one strand shall be removed from one of the modified tendons in the dome, hoop and vertical directions. Portions of these strands will be tensile tested and an examination will be conducted over their entire length to determine if evidence of corrosion or other deleterious effects are present. At each successive inspection the sample selection will be rotated among the 9 modified tendons.

Should the inspection of one of the strands reveal any significant corrosion (pitting or loss of area), further inspection of the other two modified tendons in that directional group will be made to determine the extent of the corrosion and its significance to the load carrying capability of the structure. If significant corrosion is observed at any position in a strand, a tensile test will be made on a specimen, including the length with corrosion representative of the maximum observed.

Tensile tests will be made on a minimum of three specimens taken from the ends and middle of each of the three strands. These specimens will be tested to the requirements of ASTM 416 for 270,000 psi ultimate strength strand.

4.4.2.4 Anchorage Surveillance

The tendon anchorage assembly hardware of all tendons inspected will be visually checked. The surrounding concrete will also be checked visually for indications of abnormal material behavior.

The anchor heads and visible portions of wedges of all tendons inspected will be checked for grease coverage and evidence of corrosion.

Samples of grease will be taken from 3 tendons that have strands removed under strand surveillance above. These grease samples will be checked by laboratory analysis for acceptance per the requirements of the grease specification included in the C12.1A supply specification for the tendon system.

Additionally, the following will be checked on the modified hoop tendons at each inspection:

A. The shims, trumpet and tendon in the anchorage area will be inspected for grease coverage and signs of corrosion.

B. The grease coverage will be noted along with the temperature to accumulate a record of grease variation versus temperature.

4.4.2.5 Reports

A report covering the results of each inspection will be prepared, reviewed by the District's Generation Engineering Department and filed with the plant quality assurance records. If any significant or critical deterioration is noted by this inspection, it will be reported to the NRC as a reportable occurrence in accordance with Technical Specification 6.9-1. Should this be necessary, the initial report may be made within 10 days of the completion of the tests and the detailed report may follow within 90 days of the completion of the tests.

4.4.2.6 Liner Plate Surveillances

The liner plate will be examined prior to the initial pressure test in accessible areas to determine the following:

a. Location of areas which have inward deformations. The magnitude of the inward deformations shall be measured and recorded. These areas shall be permanently marked for future reference and the inward deformations shall be measured between the angle stiffeners which are on 15" centers. The measurements shall be accurate to 0.01". Temperature readings shall be obtained on both the liner plate and outside containment wall at the locations where inward deformations occur.

b. Locations of areas having strain concentrations by visual examination with emphasis on the condition of the liner surface. The location of these areas shall be recorded.

4.4.2.6.2 Shortly after the initial pressure test and approximately one year after initial startup, a re-examination of the areas located in paragraph 4.4.2.6.1 A shall be made. Measurements of the inward deformations and observations of any strain concentrations shall be made.

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

4.4.2.6.4 The surveillance program shall be discontinued after the one year initial startup inspection if no corrective action was needed. If corrective action is required, the frequency of inspection for a continued surveillance program shall be determined.

4.4.2 (Continued)

Bases

Provisions have been made for an inservice surveillance program, covering the first five years of the life of the unit, intended to provide sufficient evidence to maintain confidence that the integrity of the reactor building is being preserved. This program consists of tendon, tendon anchorage and liner plate surveillance.

To accomplish these programs, two separate sets of nine tendons each are used. Each of the sets consists of three horizontal tendons, three vertical tendons and three dome tendons. The locations of these 18 tendons are shown in Figure 5A-21 of the FSAR.

In its normal configuration, the VSL wedge anchored strand tendon system cannot be detensioned without destroying the tendon. The anchorages of three hoop tendons have been modified by the addition of shims to permit them to be detensioned. The shims are placed between the bearing plate and the anchor head prior to initial tensioning and are of a total length at least equal to the tendon elongation. During surveillance, these shims are removed in increments until the tendon is detensioned. Modified dome and vertical tendons have additional length extending beyond the anchor head to facilitate removal of a corrosion surveillance strand.

Strand continuity cannot be checked by pulling each strand to observe its movement at the opposite end since the wedges are held in the anchor head by a residual clamping force after the tendon is completely detensioned. The wedges should not be dislodged since it is not advisable to regrip the strand in the same place.

The inspection during this initial five year period of at least one strand from each of the nine corrosion surveillance tendons is considered sufficient representation to detect the presence of any widespread tendon corrosion or pitting conditions in the structure. This program will be subject to review and revision as warranted based on studies and on results obtained for this and other prestressed concrete reactor buildings during this period of time.

REFERENCE

FSAR paragraph 5.2.5.3

4.4.3 Hydrogen Purge System

Applicability

Applies to testing Reactor Building hydrogen purge system.

Objective

To verify that this system and components are operable.

Specification

4.4.3.1 Operating Tests

An in-place system test shall be performed during each refueling interval. These tests shall consist of visual inspection and a flow measurement using the installed flow instruments. Flow shall be design flow or higher. Blower motors shall be operated continuously for at least one hour and valves shall be proven operable.

4.4.3.2 H_2 Detector Test

The hydrogen concentration analyzer shall be calibrated yearly.

Bases

The hydrogen purge system is composed of permanently installed, independent, centrifugal exhaust blower, portions of the Reactor Building radiation monitoring and sampling system and the waste gas system. This controlled purge system is completely independent of the larger purge system employed, as required, during normal plant operation prior to building entry. Since this system is not normally operated, a periodic test is required to insure its operability when needed. During this test, the system will be inspected for such things as water, oil or other foreign material, gasket deterioration, and unusual or excessive noise or vibration when the blowers are running.

4.5 EMERGENCY CORE COOLING AND REACTOR BUILDING COOLING SYSTEM PERIODIC TESTING

4.5.1 Emergency Core Cooling System

Applicability

Applies to periodic testing requirement for emergency core cooling systems.

Objective

To verify that the emergency core cooling systems are operable.

Specification

4.5.1.1 Systems Tests

A. High Pressure Injection

1. During each refueling interval, a makeup and purification system test shall be conducted to demonstrate the system is operable for high pressure injection. A manual trip signal will be applied to demonstrate actuation of the makeup and purification for emergency core cooling operation.

2. The test will be considered satisfactory if control board indication verifies that all components have responded to the actuation signal; all appropriate pump breakers shall have opened or closed and all valves have completed their travel.

3. The high pressure injection pump casings shall be vented monthly and prior to any ECCS flow tests.

B. Low Pressure Injection

1. During each refueling interval a decay heat removal system test shall be conducted to demonstrate the system is operable for low pressure injection. The test shall be performed in accordance with the procedure summarized below:

   a. A manual trip signal will be applied to demonstrate actuation of the decay heat removal system for emergency core cooling operation.

   b. Verification of the safety features function of the nuclear service cooling water system and nuclear service raw water system which supplies cooling water to the decay heat removal coolers shall be made to demonstrate operability of the coolers.

4.5.1.1 B. (Continued)

2. The test will be considered satisfactory if control board indication verifies that all components have responded to the actuation signal and all appropriate pump breakers shall have opened or closed, and all valves have completed their travel.

3. Decay heat pump casing shall be vented monthly and prior to any ECCS flow tests.

4. Periodic leakage testing[a] on each valve listed in Table 3.3-1 shall be accomplished prior to plant operation at power after every time the plant is placed in the cold shutdown condition for refueling, after each time the plant is placed in a cold shutdown condition for 72 hours if testing has not been accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance, repair or replacement work is performed.

5. Whenever integrity of a pressure isolation valve listed in Table 3.3-1 cannot be demonstrated, the integrity of the remaining valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of the other closed valve located in the high pressure piping shall be recorded daily.

C. Core Flooding System

1. During each refueling interval, a core flooding system test shall be conducted to demonstrate proper operation of the system. During depressurization of the reactor coolant system, verification shall be made that the check valves in the core flooding tank discharge lines operate.

2. The test will be considered satisfactory if control board indication of core flood tank level verifies that all check valves have opened.

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[a]To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

4.5.1.2 Components Tests

A. Pumps

At least quarterly, the high pressure, makeup, and decay heat removal pumps shall be started and operated to verify operation.

B. Valves - Power Operated

At least quarterly, each safety features valve in the emergency core cooling systems and each safety features valve associated with emergency core cooling in the decay heat removal system shall be tested to verify operability.

C. Nuclear Service Cooling and Raw Water System

At least quarterly, the nuclear service cooling and raw water system shall be operated to verify performance.

D. Acceptance

1. Acceptable performance for the high pressure injection pumps shall be that the pump starts and operates for 15 minutes discharging through the miniflow and the discharge pressure indicates flow is within 10% of the initial level of performance and at least equal to the minimum design flow rate.

2. Acceptable performance of the decay heat pumps shall be that the pump starts and operates for 15 minutes discharging through the test flow path and the discharge pressure and flow are within 10% of the initial level of performance and at least equal to the minimum design flow rate.

3. Acceptable performance for the nuclear service cooling water and raw water pumps shall be that the pump starts and operates for 15 minutes at the design flow rate with the required differential pressure.

4. The acceptable performance of each power-operated valve in the emergency core cooling system will be that motion is indicated upon actuation by appropriate signals.

4.5.1 (Continued)

Bases (Continued)

The emergency core cooling systems are the principal reactor safeguards in the event of a loss-of-coolant accident. The removal of heat from the core provided by these systems is designed to limit core damage.

The decay heat removal pumps are tested singularly for operability by opening the borated water storage tank outlet valves and the test line valves to the borated water storage tank. This allows water to be pumped from the borated water storage tank through each of the injection lines and back to the tank through a test line.

With the reactor shut down, the check valves in each core flooding line are checked for operability by reducing the reactor coolant system pressure until the indicated level in the core flood tanks verify the check valves have opened.

REFERENCES

FSAR subsection 6.2

4.5.2 Reactor Building Cooling Systems

Applicability

Applies to testing of the Reactor Building cooling system.

Objective

To verify that the Reactor Building cooling systems are operable.

Specification

4.5.2.1 System Tests

A. Reactor Building Spray System

1. During each refueling interval a system test shall be conducted to demonstrate proper operation of the system. A manual trip signal will be applied to demonstrate actuation of the Reactor Building spray system (except for the Reactor Building motor-operated inlet valves which prevent water entering nozzles). Water will be circulated from the borated water storage tank through the Reactor Building spray pumps and returned through the test line to the borated water storage tank.

2. Air will be introduced into the spray headers to verify the availability of the headers and spray nozzle at least every 10 years.

3. The test will be considered satisfactory if visual observation and control board indication verifies that all components have responded to the actuation signal and the appropriate pump breakers shall have opened and closed, and all valves shall have completed their travel except the blocked Reactor Building inlet valve.

B. Reactor Building Emergency Cooling System

1. During each refueling interval, a system test shall be conducted to demonstrate proper operation of the system, including the upper dome air circulators. The test shall be performed in accordance with the procedure summarized below:

   a. A manual trip signal will be applied to actuate the Reactor Building emergency cooling system for Reactor Building cooling operation.

   b. Verification of the safety features function of the nuclear service cooling water system which supplies coolant water to the Reactor Building coolers shall be made to demonstrate the operability of the coolers.

4.5.2.1 (Continued)

2. The test will be considered satisfactory if control board indication verifies that all components have responded to the actuation signal and the appropriate pump breakers shall have completed their travel. NSCW flow through each operating cooler exceeds 1400 gpm, fans are running and air flow through each cooler exceeds 40,000 cfm.

4.5.2.2 Component Tests

A. Pumps

At least quarterly, the Reactor Building spray pumps shall be started and operated to verify proper operation. Acceptable performance will be indicated if the pump starts, operates for 15 minutes and the discharge pressure and flow are within 10% of the initial level of performance and at least equal to the minimum design flow rate.

B. Valves

At quarterly intervals each safety features valve in the Reactor Building spray and Reactor Building emergency cooling system and each safety features valve associated with the nuclear service water cooling system shall be tested to verify that it is operable.

Bases

The Reactor Building emergency cooling systems and Reactor Building spray system are designed to remove the heat in the containment atmosphere to prevent the building pressure from exceeding the design pressure.[1]

The delivery capability of one Reactor Building spray pump at a time can be tested by opening the valve in the line from the borated water storage tank, opening the corresponding valve in the test line and starting the corresponding pump. Pump discharge pressure and flow indication demonstrate performance.

With the pumps shut down and the borated water storage tank outlet valves closed, the Reactor Building spray injection valves can each be opened and closed by operator action. With the Reactor Building spray inlet valves closed, air can be blown through the test connections of the Reactor Building spray nozzles to demonstrate that the flow paths are open.

4.5.2 (Continued)

Bases (Continued)

The equipment, piping, valves and instrumentation of the Reactor Building emergency cooling system are arranged so that they can be visually inspected. The cooling units and associated piping are located outside the secondary concrete shield. Personnel can enter the Reactor Building during power operations to inspect and maintain this equipment. The nuclear service cooling water piping and valves outside the Reactor Building are inspectable at all times. Operational tests shall be performed prior to initial startup.

REFERENCES

FSAR section 9

4.5.3 Decay Heat Removal System Leakage

Applicability

Applies to decay heat removal system leakage.

Objective

To maintain a preventative leakage rate for the decay heat removal system which will prevent significant offsite exposures.

Specification

4.5.3.1 Acceptance Limit

The maximum allowable leakage from the decay heat removal system components (which includes valve stems, flanges and pump seals) shall not exceed 0.63 gallons per hour.

4.5.3.2 Test

During each refueling interval the following tests of the decay heat removal system shall be conducted to determine leakage:

A. The portion of the decay heat removal system, except as specified in (B), that is outside the containment shall be tested either by use in normal operation or by hydrostatically testing at 450 psig.

B. Piping from the containment emergency sump to the decay heat removal pump suction isolation valve shall be pressure tested at no less than 52 psig as a containment local leak rate test under paragraph 4.4.1.2.

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

Bases

The leakage rate limit for the decay heat removal system is a judgement value based on assuring that the components can be expected to operate without mechanical failure for a period on the order of 200 days after a loss of coolant accident. The test pressure (450 psig) achieved either by normal system operation of by hydrostatically testing, gives an adequate margin over the highest pressure within the system after a design basis accident. Similarly, the pressure test for the return lines from the containment to the decay heat removal system (52 psig) is equivalent to the peak calculated pressure after a LOCA. A decay heat removal system leakage of 0.63 gal/h will

4.5.3.2 (Continued)

Bases (Continued)

limit off-site exposures due to leakage to insignificant levels relative to those calculated for leakage directly from the Reactor Building in the design basis accident. The dose to the thyroid calculated as a result of this leakage is 0.76 rem for a two hour exposure at the site boundary. (1)

REFERENCES

(1) FSAR paragraph 14.3.9.3

4.6 EMERGENCY POWER SYSTEM PERIODIC TESTING

Applicability

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

Objective

To verify that the emergency power sources and equipment are operable and respond properly when required.

Specification

4.6.1 At intervals not to exceed one month, a test of the diesel generators will be performed to verify proper operation of these emergency power sources and associated equipment. This test will be performed to assure that:

A. Each diesel generator can be started from the control room.

B. Each diesel generator can be synchronized with its associated 4160 volt nuclear service bus.

4.6.2 During each refueling interval, a test of the diesel generators and emergency start circuits shall be performed to verify that these emergency power sources and associated equipment are available to carry load within 15 seconds of a simulated requirement for the safety features system.

A. Start diesel generator by SFAS signal

B. Test the diesel generator sequencing circuits for SFAS load pickup.

C. Load test diesel generators to SFS capacity.

4.6.3 Each diesel generator shall be given a thorough inspection at least biennually following the manufacturer's recommendations for this class of standby service.

4.6.4 Batteries in the 125 volt d-c systems shall be tested as follows:

A. The voltage and specific gravity of each pilot cell shall be measured and recorded weekly.

B. The specific gravity, level and voltage of each cell shall be measured and recorded every month.

4.6.4 (Continued)

C. Each time data are recorded, new data shall be compared with old to detect signs of deterioration.

D. During each refueling interval, the battery shall be subjected to a rated load or equivalent test. The battery voltage as a function of time shall be monitored to establish that the battery performs as expected.

4.6.5 Diesel generator fuel oil supply shall be tested as follows:

A. During the monthly diesel generator test, the diesel fuel oil transfer pumps shall be monitored for operation.

B. Once a month, quantity of the diesel fuel oil shall be logged and checked against minimum specifications.

The tests specified will be considered satisfactory if control room indication and/or visual examination demonstrate that all components have operated properly.

4.6.6 The pressurizer shall be tested as follows:

A. The pressurizer water level shall be determined to be within its limits at least once per 12 hours.

B The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by transferring power from the normal to the emergency power supply and energizing the heaters.

Bases

The tests specified are designed to demonstrate that the diesel generators will provide power for operation of safety features equipment. They also assure that the emergency generator control system and the control systems for the safety features equipment will function automatically in the event of a loss of all normal a-c station service power or upon the receipt of a safety features actuation signal. The testing frequency specified is intended to identify and permit correction of any mechanical or electrical deficiency before it can result in a system failure. The fuel oil supply, starting circuits and controls are continuously monitored and any faults are alarmed and indicated. An abnormal condition in these systems would be signaled without having to place the diesel generators on test.

Precipitous failure of the plant battery 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.

REFERENCE

(1) IEEE 308

4.7 REACTOR CONTROL ROD SYSTEM TESTS

4.7.1 Control Rod Drive System Functional Tests

Applicability

Applies to the surveillance of the control rod system.

Objective

To assure operability of the control rod system.

Specification

4.7.1.1 The control rod trip insertion time shall be measured for each control rod at either full flow or no flow conditions following each refueling outage prior to return to power. The maximum control rod trip insertion time for an operable control rod drive mechanism, except for the axial power shaping rods (APSRs), from the fully withdrawn position to 3/4 insertion (104" travel) shall not exceed 1.66 seconds at hot, reactor coolant full flow conditions or 1.40 seconds for hot, no flow conditions. For the APSRs it shall be demonstrated that loss of power will not cause rod movement. If the trip insertion time above is not met, the rod shall be declared inoperable.

4.7.1.2 If a control rod is misaligned with its group average by more than an indicated nine inches, the rod shall be declared inoperable and the limits of Specification 3.5.2.2 shall apply. The rod with the greatest misalignment shall be evaluated first. The position of a rod declared inoperable due to misalignment shall not be included in computing the average position of the group for determining the operability of rods with lesser misalignments.

4.7.1.3 If a control rod cannot be exercised, or if it cannot be located with absolute or relative position indications or in or out limit lights, the rod shall be declared to be inoperable.

Bases

The control rod trip insertion time is the total elapsed time from power interruption at the control rod drive breakers until the control rod has completed 104" of travel from the fully withdrawn position. The specified trip time is based upon the safety analysis in FSAR section 14.

Each control rod drive mechanism shall be exercised by a movement of approximately two inches of travel every two weeks. This requirement shall apply to either a partial or fully withdrawn control rod at reactor operating conditions. Exercising the drive mechanisms in this manner provides assurance of reliability of the mechanisms.

4.7.2 Control Rod Program Verification (Group vs Core Positions)

Applicability

Applies to surveillance of the control rod systems.

Objective

To verify that the designated control rod (by core position 1 through 69) is operating in its programmed functional position and group. (rod 1 through 12, group 1-8)

Specification

4.7.2.1 Whenever the control rod drive patch panel is locked (after inspection, test, reprogramming, or maintenance), each control rod drive mechanism shall be selected from the control room and exercised by a movement of not more than two inches to verify that the proper rod has responded as shown on the unit computer printout of that rod or on the input to the computer for that rod.

4.7.2.2 Whenever power or instrumentation cables to the control rod drive assemblies atop the reactor or at the bulkhead are disconnected or removed, an independent verification check of their reconnection shall be performed.

4.7.2.3 Any rod found to be improperly programmed shall be declared inoperable until properly programmed.

Bases

Each control rod has a relative and an absolute position indicator system. One set of outputs goes to the plant computer identified by a unique number (1 through 69) associated with only one core position. The other set of outputs goes to a programmable bank of 69 edgewise meters in the control room. In the event that a patching error is made in the patch panel or connectors in the cables leading to the control rod drive assemblies or to the control room meter bank are improperly transposed upon reconnection, these errors and transpositions will be discovered by a comparative check by (1) selecting a specific rod from one group (e.g. rod 1 in regulating group 6) (2) noting that the program-approved core position for this rod of the group (assume the approved core position is No.53) (3) exercise the selected rod and (4) note that (a) the computer prints out both absolute and relative position response for the approved core position (assumed to be No. 53) (b) the proper meter in the control room display bank (assumed rod 1 in group 6) in both absolute and relative meter positions.

4.7.2 (Continued)

Bases (Continued)

This type of comparative check will not assure detection of improperly connected cables inside the Reactor Building. For these, (paragraph 4.7.2.2) it will be necessary for a responsible person, other than the one doing the work, to verify by appropriate means that each cable has been matched to the proper control rod drive assembly.

A rod is considered inoperable if it cannot be exercised, if the trip insertion time is greater than the specified allowable time, or if the rod deviates from its group average position by more than nine inches. Conditions for operation with an inoperable rod are specified in Technical Specification 3.5.2.

REFERENCES

FSAR section 14

4.8 AUXILIARY FEEDWATER PUMP PERIODIC TESTING

Applicability

Applies to the periodic testing of the turbine and motor driven auxiliary feedwater pumps.

Objective

To verify that the auxiliary feedwater pump and associated valves are operable.

Specification

4.8.1 At least every 92 days at a time when the average reactor coolant system temperature is $\geq$305 F, the turbine/motor driven and motor driven auxiliary feedwater pumps shall be operated on recirculation to the condenser to verify proper operation.

The 92 day test frequency requirement shall be brought current within 72 hours after the average reactor coolant system temperature is $\geq$305 F.

Acceptable performance will be indicated if the pump starts and operates for 15 minutes at the design flow of 780 gpm. This flow will be verified using tank level decrease and pump differential pressure.

4.8.2 At least once per 18 months during a shutdown:

1. Verify that each automatic valve in the flow path actuates to its correct position upon receipt of each auxiliary feedwater actuation test signal.

2. Verify that each auxiliary feedwater pump starts as designed automatically upon receipt of each auxiliary feedwater actuation test signal.

4.8.3 All valves, including those that are locked, sealed, or otherwise secured in position, are to be inspected monthly to verify they are in the proper position.

4.8.4 Prior to startup following a refueling shutdown or any cold shutdown of longer than 30 days duration, conduct a test to demonstrate that the motor-driven AFW pumps can pump water from the CST to the steam generator.

4.8 (Continued)

Bases (Continued)

The quarterly test frequency will be sufficient to verify that the turbine/motor driven and motor driven auxiliary feedwater pumps are operable. Verification of correct operation will be made both from the control room instrumentation and direct visual observation of the pumps.

The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than 305 F from normal operating conditions in the event of a total loss of off-site power.

Each electric driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 780 gpm at a pressure of 1050 psig to the entrance of the steam generators. The steam driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 780 gpm at a pressure of 1050 psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 300 F when the Decay Heat Removal System may be placed into operation.

4.9 REACTIVITY ANOMALIES

Applicability

Applies to potential reactivity anomalies.

Objective

To require the evaluation of reactivity anomalies of a specified magnitude occurring during the operation of the unit.

Specification

Following a normalization of the computed boron concentration as a function of burnup, the actual boron concentration of the coolant shall be monthly compared with the predicted value. If the difference between the observed and predicted steady-state concentrations reaches the equivalent of one percent in reactivity, an evaluation will be made to determine the cause of the discrepancy and reported to the Nuclear Regulatory Commission.

Bases

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 percent 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 that 1 percent would be unexpected and its occurrence would be thoroughly investigated and evaluated.

The value of one percent is considered a safe limit since a shutdown margin of at least one percent with the most reactive rod in the fully withdrawn position is always maintained.

4.10 EMERGENCY CONTROL ROOM FILTERING SYSTEM

Applicability

Applies to the emergency control room filtering system components.

Objective

To verify that these systems and components will be able to perform their design functions.

Specification

4.10.1 Operating Tests

System tests shall be performed at approximately quarterly intervals. These tests shall consist of visual inspection, a flow measurement using a flow instrument installed at the outlet of each filter unit and pressure drop measurements across the filter bank. Pressure drop across pre-filter shall not exceed one inch H_2O and pressure drop across HEPA shall not exceed two inches H_2O. Fan motors shall be operated continuously for at least one hour and all louvers and other mechanical systems shall be proven operable.

4.10.2 Filter Tests

During each refueling interval an "in-place" leakage test using DOP on HEPA units and Freon-112 (or equivalent) on the charcoal units shall be performed at filter train design flow. Removal of 99.9 percent DOP by each entire HEPA filter unit and removal of 99.5 percent Freon-112 (or equivalent) by the charcoal adsorber unit shall constitute acceptable performance. These tests must also be performed after any maintenance which may affect the structural integrity of either the filtration system units or of the housing.

Bases

The purpose of the emergency control room filtering system is to limit the particulate and gaseous fission products to which the control area would be subjected during an accidental radioactive release in or near the Auxiliary Building. The system is designed with a filter train which consists of a prefilter, high efficiency particulate filter, charcoal filter and a booster fan to pressurize the control room with outside air.

Since this system is not normally operated, a periodic test is required to insure its operability when needed. Quarterly testing of this system will show that the system is available for its safety action. During this test the system will be inspected for such things as water, oil or other foreign material, gasket deterioration, adhesive deterioration in the HEPA units and unusual or excessive noise or vibration when the fan motors are running.

Annual testing will verify the efficiency of the charcoal and absolute filters.

4.13 AUGMENTED INSERVICE INSPECTION PROGRAM FOR HIGH ENERGY LINES OUTSIDE OF CONTAINMENT

Applicability

Applies to welds in piping systems or portions of systems located outside of containment where protection from the consequences of postulated ruptures is not provided by a system of pipe whip restraints, jet impingement barriers, protective enclosures and/or other measures designed specifically to cope with such ruptures.

For Rancho Seco Unit 1 this specification applies to welds in the main steam and main feedwater lines within the region outlined in Figures 4.13-1, 4.13-2 and 4.13-3.

Objective

To provide assurance of the continued integrity of the piping systems over their service lifetime.

Specifications

A. For the 41 welds identified on Figures 4.13-1, 4.13-2 and 4.13-3:

1. Prior to initial power operation (greater than 5 percent) a volumetric examination will be performed with 100 percent inspection of welds in accordance with the requirement of ASME Section XI Code, Inservice Inspection of Nuclear Power Plant Components, to establish system integrity and baseline data.

2. The in-service inspection at each weld will be performed in accordance with the requirements of ASME Section XI Code, Inservice Inspection of Nuclear Power Plant Components, with the following schedule: (the inspection intervals identified below sequentially follow the baseline examination of 4.13 A 1 above):

First 10 Year Inspection Program Intervals	
a. First 3-1/3 years (or nearest refueling outage)	100 percent volumetric inspection of all welds
b. Second 3-1/3 years (or nearest refueling outage)	100 percent volumetric inspection of all welds
c. Third 3-1/3 years (or nearest refueling outage)	100 percent volumetric inspection of all welds

4.13 (Continued)

<u>Successive Inspection Intervals</u>

Every 10 years thereafter (or nearest refueling outage)	Volumetric inspection of 1/3 of the welds at the expiration of each 1/3 of the inspection interval with a cummulative 100 percent coverage of all welds

Note -- The welds selected during each inspection period shall be distributed among the total number to be examined to provide a representative sampling of the conditions of the welds.

3. Examinations that reveal unacceptable structural defects in a weld during an inspection under 4.13 A 2 shall be extended to require an additional inspection of another 1/3 of the welds. If further unacceptable defects are detected in the second sampling, the remainder of the welds shall be inspected.

4. In the event repairs of any welds are required following any examination during successive inspection intervals, the inspection schedule for the repaired welds will revert back to the first 10 year inspection program.

B. For all welds in critical areas other than those identified as postulated break location on Figure 4.13-1, 2 and 3:

1. Welds in the main steam lines including the safety valve headers and in the feedwater lines including branch lines will be examined in accordance with the requirements of subsection ISC 100 through 600 of the 1972 Winter Addenda of the ASME Section XI Code.

C. For all welds in the critical areas as identified on Figure 4.13-1, 2 & 3:

1. A visual inspection of the surface of the insulation at all weld locations shall be performed on a weekly basis for detection of leaks. Any detected leaks shall be investigated and evaluated. If the leakage is caused by a through-wall flaw, either the plant shall be shutdown or the leaking piping isolated. Repairs shall be performed prior to return of this line to service.

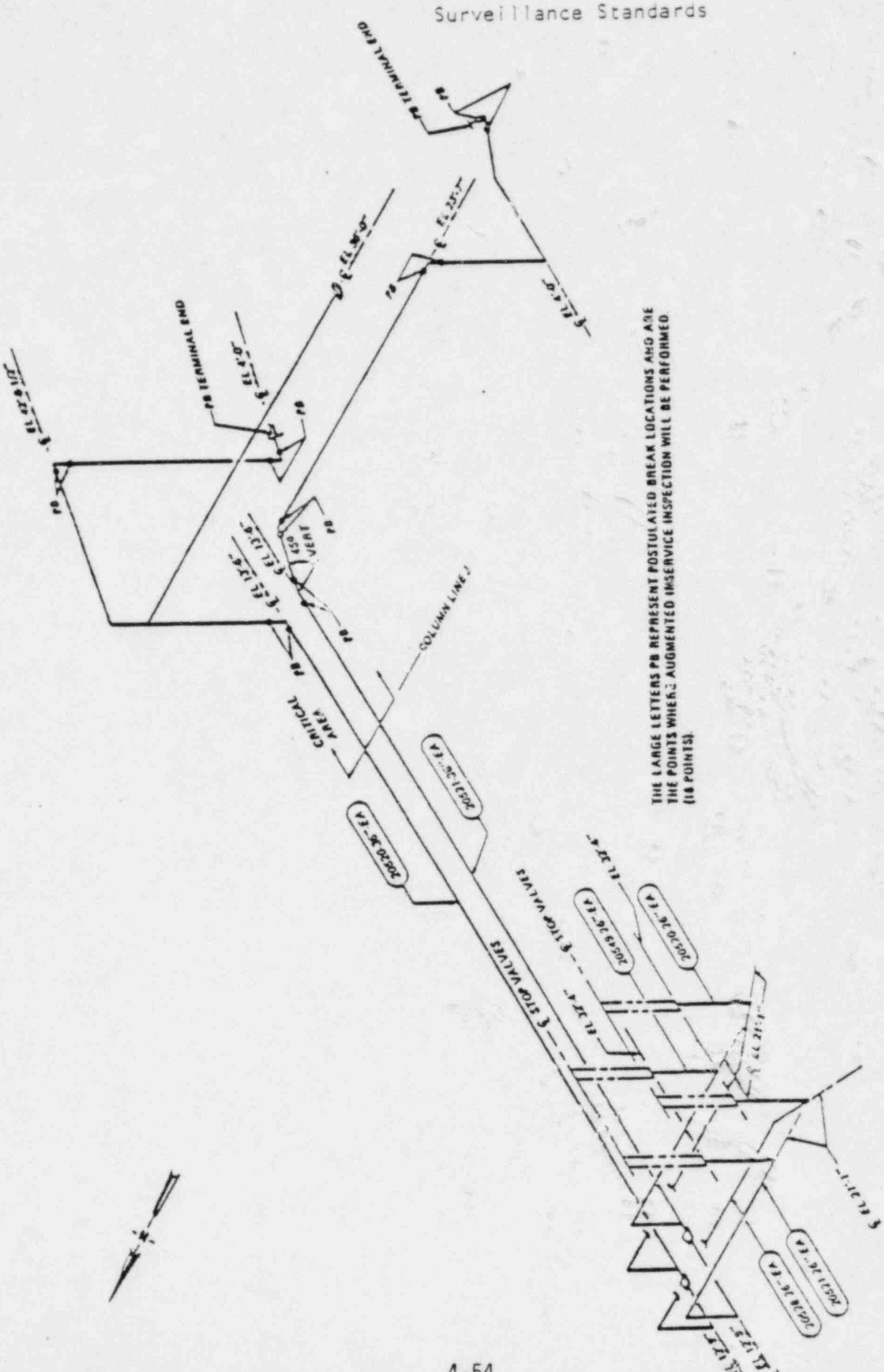
2. Repairs, re-examination and piping pressure tests shall be conducted in accordance with the rules of ASME Section XI Code.

Surveillance Standards

4.13 (Continued)

Bases

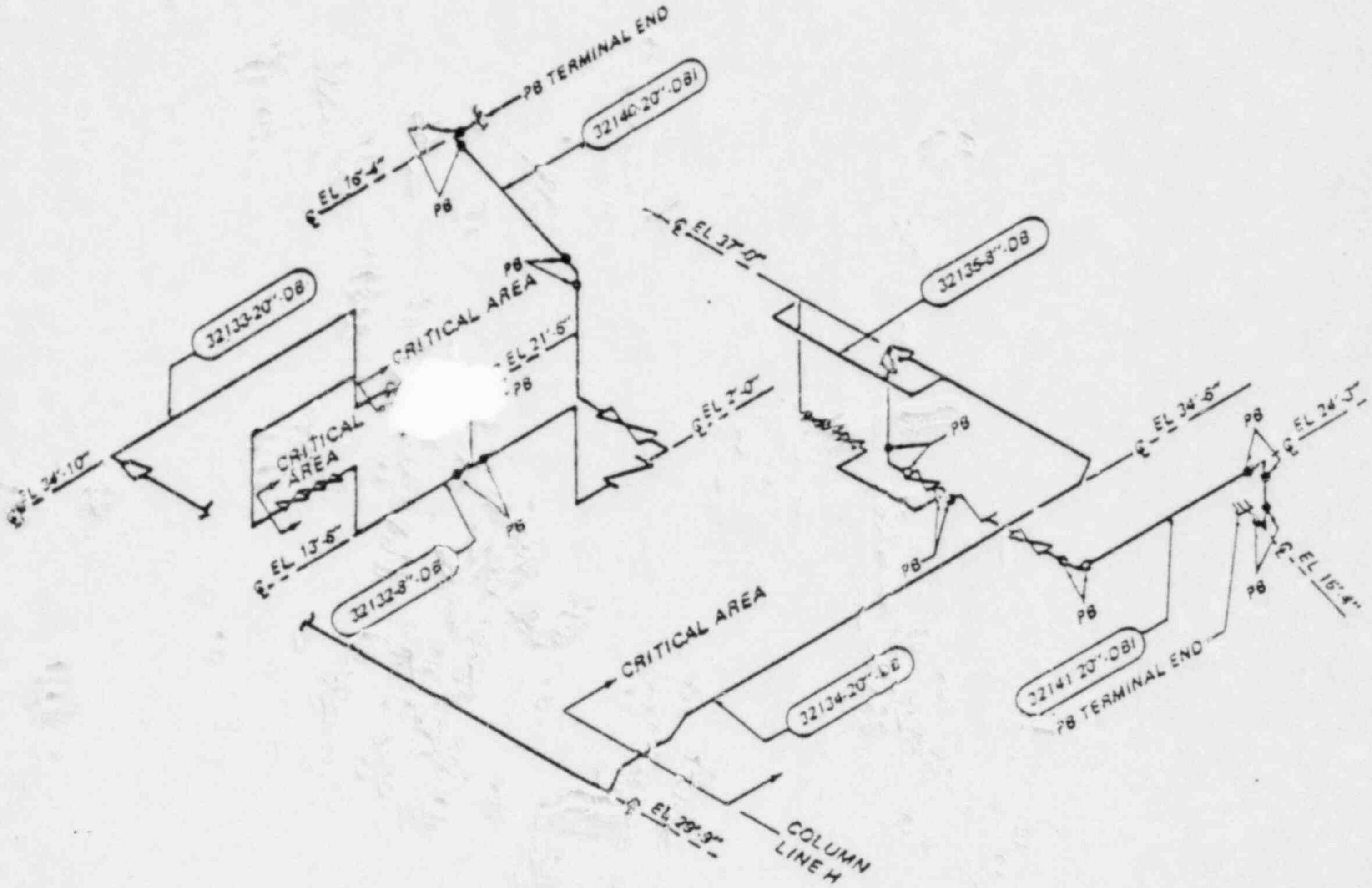
Under normal plant operating conditions, the piping materials operate under ductile conditions and within the stress limits considerably below the ultimate strength properties of the materials. Flaws which could grow under such conditions are generally associated with cyclic loads that fatigue the metal, and lead to leakage cracks. The inservice examination and the frequency of inspection will provide a means for timely detection even before the flaw penetrates the wall of the piping.



THE LARGE LETTERS PB REPRESENT POSTULATED BREAK LOCATIONS AND ARE THE POINTS WHERE AUGMENTED INSERVICE INSPECTION WILL BE PERFORMED (16 POINTS).

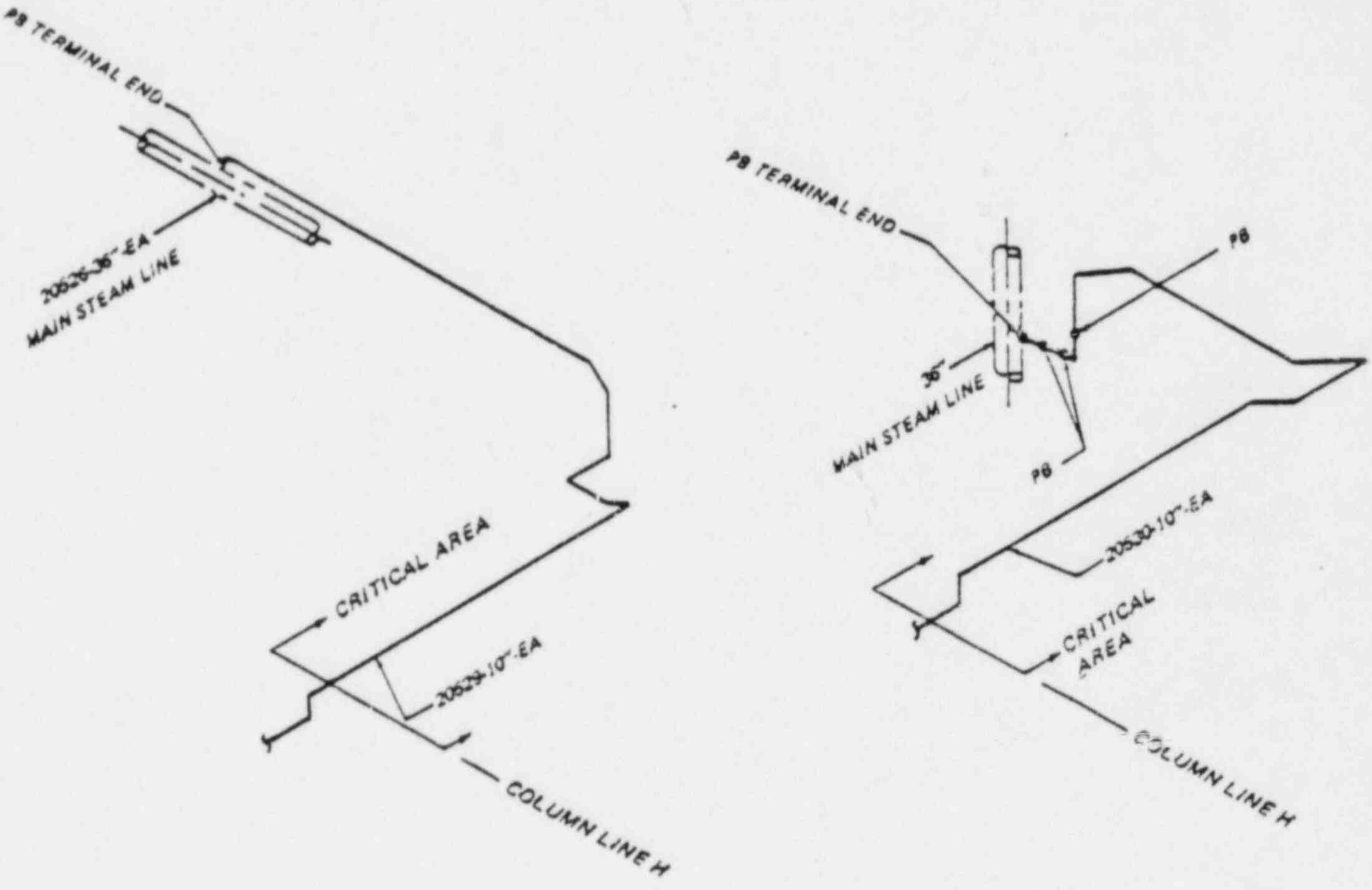
FIGURE 4.13-1
MAIN STEAM INSERVICE INSPECTION

FIGURE 4.13-2
MAIN FEEDWATER
INSERVICE INSPECTION



THE LARGE LETTERS PB REPRESENT POSTULATED BREAK LOCATIONS AND ARE THE POINTS WHERE AUGMENTED INSERVICE INSPECTION WILL BE PERFORMED. (20 POINTS).

FIGURE 4.13-3
MAIN STEAM DUMP
INSERVICE INSPECTION



THE LARGE LETTERS PB REPRESENT POSTULATED BREAK LOCATIONS AND ARE THE POINTS WHERE AUGMENTED INSERVICE INSPECTION WILL BE PERFORMED. (5 POINTS).

4.14 SHOCK SUPPRESSORS (SNUBBERS)

Applicability

The following surveillance requirements apply as applicable to hydraulic snubbers listed in Table 4.14-1.

Objective

To verify that the hydraulic snubbers will perform their design function.

Specification

4.14.1. All hydraulic snubbers whose seal material has been demonstrated by operating experience, laboratory testing or analysis to be compatible with the operating environment shall be visually inspected. This inspection shall include inspection of the hydraulic fluid reservoir level, fluid connections and linkage connections to the piping and anchor for integrity in accordance with the following schedule:

Number of Snubbers Found Inoperable During Inspection Or During Inspection Interval	Next Required Inspection Interval	
0	18 months	25%
1	12 months	25%
2	6 months	25%
3,4	4 months	25%
5,6,7	2 months	25%
8	1 month	25%

The required inspection interval shall not be lengthened more than one step at a time.

Snubbers may be categorized in two groups, "accessible" or "inaccessible" based on their accessibility for inspection during reactor operation. These two groups may be inspected independently according to the above schedule.

4.14.2. All hydraulic snubbers whose seal materials are other than ethylene propylene or material that has been demonstrated to be compatible with the operating environment, shall be visually inspected for operability monthly.

4.14.3. The initial inspection shall be performed within 6 months from the date of issuance of these specifications. For the purpose of entering the schedule in Specification 4.14.1, it shall be assumed that the facility had been on a 6 month inspection interval.

4.14.4. Once each refueling cycle, a representative sample considering different brand makes, different radiation levels, temperatures, etc., of 10 hydraulic snubbers or approximately 10% of the hydraulic snubbers, whichever is less, shall be functionally tested for operability including verification of proper piston movement, lock up and bleed. For each unit and subsequent unit found inoperable, an additional 10% or ten hydraulic snubbers shall be so tested until no more failures are found or all units have been tested. However, if all failures are shown to be unique to one brand make, and the NRC concurs in writing with this determination before reactor restart, snubbers of other brand makes need not be tested during the follow-on tests. Snubbers of rated capacity greater than 50,000 lbs. and snubbers designated in Table 4.14-1 as especially difficult to remove or in high radiation area during shutdown need not be functionally tested.

Bases

Hydraulic Snubbers

All safety related hydraulic snubbers are visually inspected for overall integrity and operability. The inspection will include verification of proper orientation, adequate hydraulic fluid level and proper attachment of snubber to piping and structures.

The inspection frequency is based upon maintaining a constant level of snubber protection. Thus the required inspection interval varies inversely with the observed snubber failures. The number of inoperable snubbers found during a required inspection determines the time interval for the next required inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

Experience at operating facilities has shown that the required surveillance program should assure an acceptable level of snubber performance provided that the seal materials are compatible with the operating environment.

Snubbers containing seal material which has not been demonstrated by operating experience, lab tests, or analysis to be compatible with the operating environment should be inspected more frequently (every month) until material compatibility is confirmed or an appropriate changeout is completed.

4.14 (Continued)

Bases (Continued)

Examination of defective snubbers at reactor facilities and material tests performed at several laboratories have shown that millable gum polyurethane deteriorates rapidly under the temperature and moisture conditions present in many snubber locations. Although molded polyurethane exhibits greater resistance to these conditions, it also may be unsuitable for application in the higher temperature environments. Data are not currently available to precisely define an upper temperature limit for the molded polyurethane. Lab tests and in-plant experience indicate that seal materials are available, primarily ethylene propylene compounds, which should give satisfactory performance under the most severe conditions expected in reactor installations.

To further increase the assurance of snubber reliability, functional tests should be performed once each refueling cycle. These tests will include stroking of the snubbers to verify proper piston movement, lock-up and bleed. Ten percent or ten snubbers, whichever is less, represents an adequate sample for such tests. Observed failures on these samples should require testing of additional units. The intent is to test all snubbers an equal number of times. Those snubbers that have a rated capacity of greater than 50,000 lbs. and snubbers designated in Table 4.14-1 as being in high radiation areas or especially difficult to remove need not be selected for functional tests provided operability was previously verified.

TABLE 4.14-1
SNUBBERS ACCESSIBLE DURING POWER OPERATIONS

List #	ID #	List #	ID #
1 xx	MS 20528 SW 1	30 xx	4 SW 20024-3A
2 xx	MS 20521 SW 11	31 xx	4 SW 26101-2
3 x	MS 20520 SW 3	32 xx	4 SW 26101-6
4 x	MS 20521 SW 13	33 xx	4 SW 25023-5A
5 xx	MS 20520 SW 5	34 xx	4 SW 26101-8A
6 x	MS 20521 SW 14	35 xx	4 SW 25023-8A
7 x	MS 20520 SW 13A	36 xx	4 SW 26100-3
8 x	MS 20520 SW 6	37 xx	4 SW 26100-1A
9 xx	MS 20526 SW 9	38 x	4 SW 26022-13A
10 xx	9 SW 30709-3A	39 xx	4 SW 26025-43
11 xx	9 SW 30708-4	40 x	4 SW 23620-3A
12 xx	9 SW 30708-5A	41 xx	4 SW 53520-3
13 xx	9 SW 30709-4	42 xx	4 SW 29101-1A
14 xx	9 SW 30709-3	101 xxx	MS 20520 SW 19
15 xx	5 SW 20530-7A	102 xxx	MS 20521 SW 20
16 xx	5 SW 20529-3A	103 xxx	MS 20521 SW 22
17 x	5 SW 53520-3	104 xxx	MS 20520 SW 21
18 xx	5 SW 32141-1A	105 xx	10 SW 20553-1A
19 x	5 SW 32141-2	106 xx	5 SW 20529-1A
20 x	10 SW 20530-3A	107 xx	5 SW 20529-4A
21 xx	12 SW 20530-3A	108 xx	5 SW 20530-8
22 x	7 SW 32120-1B	109 xx	7 SW 30800-6
23 xx	7 SW 32120-1A	110 xx	7 SW 30800-8
24 xx	7 SW 32120-1C	111 x	7 SW 30800-12
25 xx	MS 20550 SW 1	112 xx	7 SW 30800-14
26 xx	MS 20558 SW 1	113 xx	7 SW 30804-1
27 xx	MS 20532 SW 1	114 xx	7 SW 30805-2
28 xx	MS 20531 SW 1	115 xx	5 SW 23803-4A
29 x	7 SW 32140-2	116 xx	4 SW 60014-4A

Modifications to this table due to changes in high radiation areas should be submitted to the NRC as part of the next license amendment.

x Designates snubbers difficult to remove or in high radiation area during shutdown.
xx Specification 4.14.4 applies to these snubbers for lockup testing.
xxx Exempt due to a rated capacity of greater than 50,000 lbs.

TABLE 4.14-2
SNUBBERS INACCESSIBLE DURING POWER OPERATIONS

List #	ID #	List #	ID #
85 xx	1 SW 10' B&W-5	125 xx	1 SW 21021-3
86 x	1 SW 10 B&W-6	126 xx	1 SW 23823-7A
87 x	1 SW 10" B&W-7	127 xx	1 SW 21025-9A
50 x	1 SW 20025-3	128 xx	1 SW 21005-8
51 x	1 SW 20025-1A	129 xx	1 SW 23822-7B
52 xx	1 SW 26020-5A	130 xx	1 SW 21028-2
53 xx	1 SW 26524-4A	131 x	1 SW 26020-6A
54 xx	1 SW 50060-8A	132 xx	1 SW 20082-12A
55 xx	1 SW 22000-9	133 xx	1 SW 26521-2A
56 xx	1 SW 26021-5A	134 x	1 SW 21025-6
57 xx	1 SW 26021-7A	135 x	1 SW 23823-13A
58 xx	1 SW 26021-11A	136 xx	1 SW 21501-20
59 xx	1 SW 50063-2A	137 xx	1 SW 21501-28
60 xx	1 SW 23823-8	138 xx	1 SW 50050-6A
61 x	1 SW 23620-15A	SS-1 xxx	SS-1
62 x	1 SW 23625-3A	SS-2 xxx	SS-2
63 x	1 SW 29122-28A	SS-3 xxx	SS-3
64 xx	1 SW 29125-7A	SS-4 xxx	SS-4
65 x	1 SW 29125-7C	SS-5 xxx	SS-5
66 xx	MS 20523 SW2	SS-6 xxx	SS-6
67 xx	MS 20521 SW17	SS-7 xxx	SS-7
68 xx	1 SW 21925-9A	SS-8 xxx	SS-8
69 xx	1 SW 50050-2A	SS-9 xxx	SS-9
70 xx	MS 20520 SW 18	SS-10 xxx	SS-10
71 xx	MS 20522 SW2	SS-11 xxx	SS-11
72 xx	1 SW 50060-3A	SS-12 xxx	SS-12
73 xx	1 SW 32140-4	SS-13 xxx	SS-13
74 xx	1 SW 32140-2	SS-14 xxx	SS-14
75 xx	1 SW 32141-4	SS-15 x	SS-15
76 xx	1 SW 32141-2	SS-16 xxx	SS-16
77 xx	1 SW 50053-2A	SS-17 x	SS-17
117 xx	1 SW 22000-12	SS-18 xxx	SS-18
118 xx	1 SW 22000-2A	SS-19 xxx	SS-19
119 xx	1 SW 22001-2	SS-20 xxx	SS-20
120 x	1 SW 21007-7	SS-21 xxx	SS-21
121 x	1 SW 21022-5	SS-22 x	SS-22
122 xx	1 SW 21006-6A	SS-23 xxx	SS-23
123 xx	1 SW 23626-1A	SS-24 xxx	SS-24
124 xx	1 SW 21022-1		

Modifications to this table due to changes in high radiation areas should be submitted to the NCR as part of the next license amendment.

x Designates snubbers difficult to remove or in a high radiation area during shutdown

xx Specification 4.14.4 applies to these snubbers for lockup testing

xxx Exempt due to a rated capacity of greater than 50,000 lbs.

4.15 RADIOACTIVE MATERIALS SOURCES

Applicability

Applies to the radioactive materials source leakage test.

Objective

To verify that the boundary materials to contain radioactive sources does not exceed allowable limits.

Specification

4.15.1 The leakage test shall be capable of detecting the presence of 0.005 microcurie of radioactive material on the test sample. If the test reveals the presence of 0.005 microcurie or more of removable contamination, it shall immediately be withdrawn from use, decontaminated, and repaired, or be disposed of in accordance with Commission regulations. Sealed sources are exempt from such leak tests when the source contains 100 microcuries or less of beta and/or gamma emitting material or 10 microcuries or less of alpha emitting material.

4.15.2 Tests for leakage and/or contamination shall be performed by the licensee or by other persons specifically authorized by the Commission or an agreement state, as follows:

a. Each sealed source, except startup sources subject to core flux, containing radioactive material, other than hydrogen 3, with a half-life greater than 30 days and in any form other than gas shall be tested for leakage and/or contamination at intervals not to exceed six months.

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

4.15.2 (Continued)

c. Startup sources shall be leak tested prior to being subjected to core flux. If any repair or maintenance is performed on the startup source seal boundary, an additional retest shall be performed.

Bases

The objective of this specification is to assure that leakage from byproduct, source, and special nuclear radioactive material sources does not exceed allowable limits.

4.16 Reserved

4.17 STEAM GENERATORS

Applicability

Applies to in-service inspection of the steam generator tubes.

Objective

To verify the operability of each steam generator and ensure the structural integrity of the tubes as part of the reactor coolant boundary.

Specification

Each steam generator shall be demonstrated OPERABLE by performance of the following augmented in-service inspection program and the requirements of Specification 1.3.

4.17.1 Steam Generator Sample Selection and Inspection

Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting steam generators as specified in Table 4.17-1.

4.17.2 Steam Generator Tube Sample Selection and Inspection

The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.17-2. The in-service inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.17.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.17.4. The tubes selected for these inpections shall include at least 3% of the total number of tubes in both steam generators and be selected on a random basis except:

a. If the experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

b. The first sample inspection during in-service inspection (subsequent to the first in-service inspection) of each steam generator shall include:

   1. All nonplugged tubes that previously had detectable wall penetrations ($>$ 20%), and

   2. Tubes in those areas where experience has indicated potential problems.

4.17.2 (Continued)

c. The second and third sample inspections during each in-service inspection may be less than a full tube inspection by concentrating (selecting at least 50% of the tubes to be inspected) the inspection on those areas of the tube sheet array and on those portions of the tubes where tubes with imperfections were previously found.

d. A tube inspection (pursuant to Specification 4.17.4.5) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection. ("Adjacent" is interpreted to mean the nearest tube capable of being inspected.) Tubes which do not permit passage of the eddy current probe will be considered as degraded tubes when classifying inspection results.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant ($>$ 10%) further wall penetrations to be included in the above percentage calculations.

4.17.3 Inspection Frequencies

The above required in-service inspections of steam generator tubes shall be performed at the following frequencies:

4.17.3 (Continued)

a. The first in-service inspection shall be performed during the first refueling outage. Subsequent in-service inspections shall be performed at intervals of not less than 12 or more than 24 calendar months after the previous inspection. If two consecutive inspections following service result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no significant additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

b. If the results of the in-service inspection of a steam generator conducted in accordance with Table 4.17-2 at 40-month intervals falls in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until a subsequent inspection meets the conditions specified in 4.17.3a and the interval can be extended to a 40-month period.

c. Additional, unscheduled in-service inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.17-2 during the shutdown subsequent to any of the following conditions:

1. Primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.10,

2. A seismic occurrence greater than the Operating Basis Earthquake,

3. A loss-of-coolant accident requiring automatic actuation of the engineered safeguards, or

4. A main steam line or feedwater line break as defined in the FSAR.

4.17.4 Acceptance Criteria

a. As used in this Specification:

1. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications of less than 20% of the nominal

4.17.4 (Continued)

tube wall thickness, if detectable, may be considered as imperfections.

2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

3. Degraded Tube means a tube containing imperfections ≥20% of the nominal wall thickness caused by degradation.

4. Defective Tube means a tube containing an imperfection ≥40% of the nominal tube wall thickness unless higher limits are shown acceptable by analysis. Defective tubes shall be plugged.

5. Tube Inspection means an inspection of the steam generator tube from the point of entry completely to the point of exit (except as noted in 4.17.2c).

b. The steam generator shall be determined OPERABLE after completing the corresponding actions required by Table 4.17.2.

4.17.5 Reports

a. Following each in-service inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission within 15 days.

b. The results of the steam generator tube in-service inspection shall be included in the Annual Operating Report for the period in which this inspection was completed. This report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 and require notification of the Commission shall be reported pursuant to Specification 6.9

4.17.5 (Continued)

prior to resumption of plant operation. The written follow-up of this report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

Bases

The Surveillance Requirements for inspection of the steam generator tubes ensure that structural integrity of this portion of the RCS will be maintained. The surveillance requirements of steam generator tubes are based on a modification of B&W - Standard Technical Specifications, dated June 1, 1976. In-service inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or in-service conditions that lead to corrosion. In-service inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that the corrective measures can be taken.

Wastage-type defects are unlikely with AVT chemistry treatment of the secondary coolant. However even if a defect should develop in service, it will be found during scheduled in-service steam generator tube examinations. Plugging will be required for defective tubes. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing in-service inspection fall into Category C-3, these results will be reported to the Commission pursuant to Specification 6.9 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection and revision of the Technical Specifications, if necessary.

TABLE 4.17-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE

INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No
No. of Steam Generators per Unit	Two
First Inservice Inspection	All
Second & Subsequent Inservice Inspection	One[1]

Table Notation:

.1 The Inservice Inspection may be limited to one steam generator on a rotating schedule encompassing 6% of the tubes in the steam generator if the results of the first or previous inspections indicate that both steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one steam generator may be found to be more sever than those in the other steam generator. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

Table 4.17-2 STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S of the Tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S of the tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S of the tubes in this S.G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of this sample	N/A	N/A
	C-3	Inspect all tubes in this S.G., plug defective tubes and inspect 2S of the tubes in the other S.G. Notification to NRC pursuant to specification 6.9	The other S.G. is C-1	None	N/A	N/A
			The other S.G. is C-2	Perform action for C-2 result of second sample	N/A	N/A
			The other S.G. is C-3	Inspect all tubes in each S.G. and plug defective tubes. Notification to NRC pursuant to specification 6.9	N/A	N/A

$S = \frac{6}{n}\%$ Where n is the number of steam generators inspected during an inspection.

4.18 FIRE SUPPRESSION SYSTEM SURVEILLANCE

4.18.1 Instrumentation

4.18.1.1 Except for fire detection instruments inaccessible during power operation, each of the fire detection instruments in Table 3.14-1 shall be demonstrated OPERABLE at least semi-annually by a CHANNEL FUNCTIONAL TEST. Fire detectors which are not accessible during plant operation shall be demonstrated OPERABLE by the performance of a CHANNEL FUNCTIONAL TEST during each COLD SHUTDOWN exceeding 24 hours unless performed in the previous 6 months.

4.18.1.2 The NFPA Standard 720 supervised circuits associated with the detector alarms for each of the fire instruments in Table 3.14-1 shall be demonstrated OPERABLE at least semi-annually.

4.18.1.3 Non-supervised circuits associated with detector alarms, between the instrument and the control room shall be demonstrated OPERABLE at least once per 31 days.

4.18.2 Water System:

4.18.2.1 The fire suppression water system shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying the contained water supply volume.

b. At least once per 31 days on a STAGGERED TEST BASIS by starting each electric motor driven pump and operating it for at least 15 minutes on recirculation flow.

c. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path is in its correct position as indicated by position instrumentation.

d. At least once per 6 months by performance of a system flush to each test fixture.

e. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

f. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence, and:

4.18.2.f (Continued)

1. Verifying that each automatic valve in the flow path actuates to its correct position,

2. Verifying that each pump develops at least 2000 gpm at a minimum pressure of 125 psig.

3. Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel, and

4. Verifying that each fire suppression pump starts (sequentially) to maintain the fire suppression water system pressure greater than or equal to 80 psig.

g. At least once per 3 years by performing a flow test of the system in accordance with Chapter 5, Section 11 of the Fire Protection Handbook, 14th Edition, published by the National Fire Protection Association.

h. The fire pump diesel engine shall be demonstrated OPERABLE:

1. At least once per 31 days by verifying:

(a) The fuel storage tank contains at least 250 gallons of fuel, and

(b) The diesel starts from ambient conditions and operates for at least 30 minutes on recirculation flow.

2. At least once per 92 days by verifying that a sample of diesel from the fuel storage tank, obtained in accordance with ASTM-D270-65, is within the acceptable limits specified in Table 1 of ASTM-D975-74 with respect to viscosity, water content and sediment.

3. At least once per 18 months, by subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service.

4.18.2 (Continued)

i. The fire pump diesel starting 24-volt battery bank and charger shall be demonstrated OPERABLE:

1. At least once per 7 days by verifying that:

(a) The electrolyte level of each battery is above the plates, and

(b) The overall battery voltage is greater than or equal to 24 volts.

2. At least once per 92 days by verifying that the specific gravity is appropriate for continued service of the battery.

3. At least once per 18 months by verifying that:

(a) The batteries cell plates and battery racks show no visual indication of physical damage or abnormal deterioration, and

(b) The battery-to-battery and terminal connections are clean, tight, free of corrosion and coated with anticorrosion material.

4.18.3 The spray and/or sprinkler systems specified in Section 3.14.3.1 shall be demonstrated to be OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path is in its correct position visually or as indicated by position instrumentation.

b. Annually by cycling each testable valve through one complete cycle of full travel.

c. Once per 18 months:

1. By performing a system functional test which includes simulated automatic actuation of the system and verifying that the automatic valves in the flow path actuate to their correct positions, and cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel; and

4.18.3.C (Continued)

2. By inspection of spray and/or sprinkler headers to verify their integrity; and

3. By inspection of each nozzle and/or sprinkler to verify no blockage.

d. At least once per three (3) years by performing a flow test through each inspector test station valve and verifying each open head spray and/or sprinkler header is unobstructed.

4.18.4 CO_2 Systems

4.18.4.1 The CO_2 systems specified in Section 3.14.4.1 shall be demonstrated OPERABLE:

a. Once per 7 days by verifying the CO_2 storage tank level to be greater than 66% and pressure to be greater than 275 psig.

b. Once per 31 days by verifying that each valve (manual, power operated, or automatic) in the flow path is in the correct position visually or as indicated by position instrumentation.

c. Once per 18 months by verifying the system electro manual pilot valves and associated ventilation dampers actuate automatically and manually to a simulated actuation signal. A flow test shall be made to verify flow from each nozzle.

4.18.5 Fire Hose Stations

4.18.5.1 Each fire hose station specified in Section 3.14.5.1 shall be demonstrated to be OPERABLE:

a. Once per 31 days by visual inspection of the station to assure all equipment is available at the station.

b. Once per 18 months inspect and replace all gaskets in the couplings that are degraded, and remove the hose for inspection and re-racking.

c. Once per three (3) years, partially open hose station valves to verify valve operability and no valve blockage.

d. Once per three (3) years by removing and replacing all hose with equivalent NFPA tested and approved hose.

4.18.6 Fire Barrier Penetrations

4.18.6.1 Each of the fire barrier penetrations specified in Section 3.14.6.1 shall be verified to be functional:

a. At least once per 18 months by a visual inspection.

b. Prior to returning a fire barrier penetration to functional status following repairs or maintenance by performance of a visual inspection of the affected fire barrier penetration(s).

4.19 RADIOACTIVE LIQUID EFFLUENT INSTRUMENTATION

Surveillance Requrements

The maximum setpoint shall be determined in accordance with procedures as described in the ODCM and shall be recorded on the release permits.

Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the INSTRUMENT CHANNEL CHECK, SOURCE CHECK, INSTRUMENT CHANNEL CALIBRATION, AND CHANNEL TEST at the frequencies shown in Table 4.19-1.

Records shall be maintained in the Process Standards of all radioactive liquid effluent monitoring instrumentation alarm/trip setpoints. Maximum setpoints and setpoint calculations shall be available for review to ensure that the limits of Specification 3.21 are met.

Bases

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual releases. The alarm/trip setpoints for these instruments shall be calculated in accordance with methods in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

TABLE 4.19-1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

Instrument	Instrument Channel Check	Source Check	Instrument Channel Calibration	Channel Test
1. Gross Beta or Gamma Radioactivity Monitors Providing Alarm and Automatic Isolation				
a. Regenerant Hold-Up Tank Discharge Line Monitor	D (1)	M (5)	R (2)	Q (3)
2. Flow Rate Monitors				
a. Waste Water Flow	D (4)	NA	R	NA

Table Notation

(1) During releases via this pathway a check shall be performed at least once per 24 hours.

(2) The Instrument Channel Calibration for radioactivity measurement instrumentation shall be performed using one or more reference standards.

(3) The Channel Test shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exist:

a. Instrument indicates measured levels above the alarm/trip setpoint.

b. Circuit failure.

c. Instrument indicates a downscale failure.

d. Instrument controls not set in operate mode.

(4) The Instrument Channel Check shall consist of verifying indication of flow during periods of release. The Instrument Channel Check shall be made at least once daily on any day on which batch releases are made.

(5) During releases via this pathway during periods of known activity in the period of primary to feedwater to the regenerant tank perform a source check daily.

4.20 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Surveillance Requirements

The maximum setpoints shall be determined in accordance with procedures as described in the ODCM and shall be recorded on release permits.

Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the INSTRUMENT CHANNEL CHECK, SOURCE CHECK, INSTRUMENT CHANNEL CALIBRATION, AND CHANNEL TEST at the frequencies shown in Table 4.20-1.

Records shall be maintained in the Process Standards of all radioactive gaseous effluent monitoring instrumentation alarm/trip setpoints. Maximum setpoints and setpoint calculations shall be available for review to ensure that the limits of Specification 3.22 are met.

Bases

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases. The alarm/trip setpoints for these instruments shall be calculated in accordance with the methods in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements and General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

The flow rates in the Reactor Building Purge Vent, Auxiliary Building Stack and Radwaste Service Area Vent are constant as they use single speed fans. The Rector Building Purge Vent has two different flow rates, winter and summer, however administrative controls assure using the correct flow rate where applicable. The actual flow rate of the ventilation systems are periodically determined by surveillance procedures. The flow rate measurement devices are used only as flow indicating devices and not for actual measurement of flow rate. Also, as these flow rate devices must be removed from the ventilation system for the channel test, and in addition transported to the manufacturer for calibration, the frequencies have been set as shown in Table 4.20-1.

Table 4.20-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

Instrument	Instrument Channel Check	Source Check	Instrument Channel Calibration	Channel Test
1. Reactor Building Purge Vent				
a. Noble Gas Activity Monitor	D(1)	M	Q(2)	Q(3)
b. Iodine Sampler	W	NA	NA	NA
c. Particulate Monitor	W	M	Q(2)	Q
d. System Effluent Flow Rate Device	W	NA	BY	A
e. Sampler Monitor Flow Rate Measurement Device	W	NA	BY	A
2. Auxiliary Building Stack				
a. Noble Gas Activity Monitor	D(1)	M	Q(2)	Q(3)
b. Iodine Sampler	W	NA	NA	NA
c. Particulate Monitor	W	M	Q(2)	Q
d. System Effluent Flow Rate Device*	W	NA	BY	A
e. Monitor Flow Rate Measurement Device	W	NA	BY	A

* This flow rate device is not yet installed. This specification for this system will become effective when it is declared OPERABLE.

Surveillance Standards

Table 4.20-1 (continued)

Instrument	Instrument Channel Check	Source Check	Instrument Channel Calibration	Channel Test
3. Radwaste Service Area*				
a. Noble Gas Activity Monitor	D (1)	M	Q (2)	Q (4)
b. Iodine Monitor	W	M	Q (2)	Q
c. Particulate Monitor	W	M	Q (2)	Q
d. System Effluent Flow Rate Device	W	NA	BY	A
e. Monitor Flow Rate Measurement Device	W	NA	BY	A

* The Radwaste Service Area Monitoring System is not yet functional. The specification for this system will become effective when it is declared OPERABLE.

Table Notation

(1) During releases via this pathway, a check shall be performed at least once per 24 hours.

(2) The Instrument Channel Calibration for radioactivity measurement instrumentation shall be performed using one or more reference standards.

(3) The Channel Test shall also demonstrate that automatic termination of this pathway and control room alarm annunciation occurs if any of the following conditions exist:

a. Instrument indicates measured levels above the alarm/strip setpoint.

b. Circuit failure.

c. Instrument indicates a downscale failure.

d. Instrument controls not set in operate mode.

Table 4.20-1 (continued)

(4) The Channel Test shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exist:

a. Instrument indicates measured levels above the alarm/trip setpoint.

b. Circuit failure.

c. Instrument indicates a downscale failure.

d. Instrument controls not set in operate mode.

4.21 LIQUID EFFLUENTS

4.21.1 Concentration

Surveillance Requirements

The concentration of radioactive material at any time in liquid effluents released from the site shall be continuously monitored in accordance with Table 3.19-1.

The liquid effluent continuous monitor having provisions for automatic termination of liquid releases, as listed in Table 3.19-1, shall be used to limit the concentration of radioactive material released at any time from the site to areas beyond the site boundary to the values given in Specification 3.21.1.

The radioactivity content of each batch of radioactive liquid waste to be discharged shall be determined prior to release by sampling and analysis in accordance with Table 4.21-1. The results of pre-release analyses shall be used with the calculational methods in the ODCM to assure that the concentration at the point of release is limited to the values in Specification 3.21.1.

Post-release analyses of samples from batch relases shall be performed in accordance with Table 4.21-1. The results of the post-release analyses shall be used with the calculational methods in the ODCM to assure that the concentrations at the point of release are limited to the values in Specification 3.21.1.

Bases

This specification is provided to ensure that the concentration of radioactive materials relased in liquid waste effluents from the site to areas beyond the site boundary will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II. This limitation provides additional assurance that the levels of radioactive materials in bodies of water outside the site will not result in exposures within: (1) the Section II.A Design Objectives of Appendix I, 10 CFR Part 50, to an individual, and (2) the limits of 10 CFR Part 10.106(e) to the population. The concentration limit for noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its MPC in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

There are no continuous releases of radioactive or potentially radioactive liquids from the plant. All releases from the plant are by batch method.

TABLE 4.21-1
RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit Of Detection (LLD) (uCi/ml)(a)
A. Batch Waste Release Tanks(b)	Each Batch P	Each Batch P	Mn-54, Fe-59, Co-58, Co-60 Zn-65, Mo-99, Cs-134, Cs-137 Ce-141, and Ce-144 (c)	5×10^{-7}
			I-131	1×10^{-6}
	One Batch/M	M	Dissolved and Entrained Gases	1×10^{-5}
	Each Batch P	M Composite(d)	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	Each Batch P	Q Composite(d)	Sr-89, Sr-90	5×10^{-8}

TABLE 4.21-1(continued)

Table Notation

a. The lower limit of detection (LLD) is defined in the ODCM.

b. A batch release is the discharge of liquid wastes of discrete volume. Prior to sampling, each batch will be isolated and thoroughly mixed, per the ODCM, to assure representative sampling.

c. Other peaks which are measureable and identifiable, together with the listed nuclides, shall also be identified and reported. Nuclides which are below the LLD for the analysis should not be reported as being present at the LLD level.

d. A composite sample is one in which the quantity of liquid samples is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen which is representative of the liquids released.

4.21.2 Doses

Dose Calculations

Cumulative dose contributions from liquid effluents shall be determined in accordance with the Offsite Dose Calculation Manual (ODCM) at least monthly.

Bases

This specification is provided to implement the requirements of Sections II.A, III.A, and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. The ACTION statements provide the required operating flexibility and, at the same time, implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents will be kept "as low as reasonably achievable." The Dose Calculations Methodology in the ODCM implements the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I is to be shown by calculational procedures based on models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977, and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

4.21.3 Liquid Holdup Tanks*

The quantity of radioactive material contained in each tank listed in Specification 3.21.3 shall be determined to be within the specified limit by analyzing a representative sample of the tank's contents at least weekly when radioactive materials are being added to the tank.

Bases

Restricting the quantity of radioactive material contained in the specified outdoor tanks provides assurance that in the event of an uncontrolled release of the contents, the concentration at the nearest surface water supply in an unrestricted area would be less than the limits of 10CFR Part 20, Appendix B, Table II, Column 2.

*Tanks included in this specification are those outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.

4.22 GASEOUS EFFLUENTS

4.22.1 Dose Rate

Surveillance Requirements

The release rate of noble gases in gaseous effluents shall be controlled by the offsite dose rate as established in Specification 3.22.1. The dose rate shall be determined in accordance with the ODCM.

The noble gas effluent continous monitors, as listed in Table 3.20-1, shall use monitor setpoints to limit offsite doses within the values established in Specification 3.22.1.

The release rate of radioactive materials, other than noble gases, in gaseous effluents shall be determined by obtaining representative samples and performing analyses in accordance with the sampling and analysis program, specified in Table 4.22-1.

The dose rate at and beyond the site boundary, due to iodine-131, tritium, and all radionuclides in particulate form with half-lives greater than 8 days released in gaseous effluents, shall be determined to be within the required limits by using the results of the sampling and analysis program, specified in Table 4.22-1, in performing the calculations of dose rate beyond the site boundary in accordance with the ODCM.

Bases

This specification is provided to ensure that the dose rate at any time at the site boundary from gaseous effluents will be within the annual dose limits of 10 CFR Part 20 for unrestricted areas. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of an individual outside the restricted area, either within or outside the site boundary, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106(b)). For individuals who may at times be within the site boundary, the occupancy of the individual will be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the restricted area boundary. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to an individual at or beyond the restricted area boundary to 500 mrem/year to the total body or to 3,000 mrem/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to an infant via cow-milk-infant pathway to 1,500 mrem/year for the nearest cow to the plant.

TABLE 4.22-1
RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (uCi/ml)[a]
A. Waste Gas Storage Tank	Each Tank Grab Sample P	Each Tank P	Principal Gamma Emitters	1×10^{-4}
			H-3	1×10^{-6}
B. Containment Purge	Each Purge Grab Sample[(e)] P	Each Purge P	Principal Gamma Emitters	1×10^{-4}
			H-3	1×10^{-6}
C. Auxiliary Building Stack, and Radwaste Service Area Vent	M(b,c) Grab Sample	M(b)	Principal Gamma Emitters	1×10^{-4}
			H-3	1×10^{-6}
D. All Release Types as listed in A,B,C above	Continuous	W(d) Charcoal Sample	I-131	1×10^{-12}
	Continuous	W(d) Particulate Sample	Principal Gamma Emitters (I-131, Others)	1×10^{-11}
	Continuous	M Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}
	Continuous	Noble Gas Monitor	Noble Gases Beta & Gamma	1×10^{-4} as Xe-133

TABLE 4.22-1 (Continued)
RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Table Notation

a. The lower limit of detection (LLD) is defined in the ODCM.

b. Analysis shall also be performed when gross beta-gamma activity analysis of reactor coolant indicates greater than 10 uCi/ml and after each 10 uCi/ml increase in the gross beta-gamma activity analysis.

c. Tritium grab samples shall be taken at least once per seven days from the ventilation exhaust from the auxiliary building stack during refueling and anytime fuel is in the spent fuel pool and the pool temperature exceeds 110 F. Below 110°F there is essentially no evaporation from this source.

d. Samples shall be changed at least weekly with analyses to be completed within 48 hours. When samples collected for less than 24 hours are analyzed, the corresponding LLD's may be increased by a factor of 10.

e. Tritium grab samples shall be taken at least daily during refueling activities.

4.22.2 Noble Gases

Dose Calculations

Cumulative air dose contributions for the quarterly or yearly period as applicable shall be determined in accordance with the Offsite Dose Calculational Manual (ODCM) at least monthly.

Bases

This specification is provided to implement the requirements of Sections II.B, III.A, and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents will be kept "as low as reasonably achievable." The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conform with the guides of Appendix I to be shown by calculational procedures based on models and data such that the actual exposure of an individual through the appropriate pathways is unlikely to be substantially underestimated. The dose calculations established in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents will be consistent with the methodolgy provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. The ODCM equations provided for determining the air doses at the site boundary will be based upon the historical average atmospheric conditions. NUREG-0133 provides methods for dose calculations consistent with Regulatory Guides 1.109 and 1.111.

Surveillance Standards

4.22.3 Iodine 131, Tritium and Radionuclides in Particulate Form

Dose Calculations

Cumulative dose contributions for the quarterly or yearly period as applicable shall be determined in accordance with the ODCM at least monthly.

Bases

This specification is provided to implement the requirements of Sections IIC, III.A, and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section II.C of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents will be kept "as low as reasonably achievable." The ODCM calculational methods specified in the surveillance requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methods approved by the NRC for calculating the doses due to the actual release rates of the subject materials are required to be consistent with the methodology provided in Regulatory Guide 1.109, "Calculating of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50 Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for radioiodines, radioactive material in particulate form and radionuclides other than noble gases are dependent on the existing radionuclide pathways to man, beyond the site boundary. The pathways which are examined in the development of these calculations are: (1) individual inhalation of airborne radionuclides, (2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, (3) deposition onto grassy areas where milk animals and meat-producing animals graze with consumption of the milk and meat by man, and (4) deposition on the ground with subsequent exposure of man.

4.23 GASEOUS RADWASTE TREATMENT

Dose Projections

Doses due to gaseous releases beyond the site boundary shall be projected at least monthly in accordance with the ODCM.

Bases

The operability of the gaseous radwaste treatment system and the ventilation exhaust treatment systems ensures that the systems will be available for use whenever gaseous effluents require treatment prior to release to the environment. The requirement that the appropriate portions of these systems be used when specified provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as reasonably achievable." This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and Design Objective Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the guide set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

4.24 GAS STORAGE TANKS

Surveillance Requirements

The quantity of radioactive material contained in each gas storage tank shall be determined to be within the limit of 3.24 at least daily when radioactive materials are being added to the tank and the Reactor Coolant System activity exceeds the limits of Specification 3.1.4.

Bases

Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to an individual at the nearest site boundary will not exceed 500 mrem. This is consistent with Standard Review Plan 15.7.1, "Waste Gas System Failure."

Calculations have shown that the reactor coolant activity must exceed the limits of Specification 3.1.4 before the storage tank activity approaches the limits of Specification 3.24.

4.25 SOLID RADIOACTIVE WASTES

Surveillance Requirements

The PROCESS CONTROL PROGRAM shall be used to verify the SOLIDIFICATION of at lease one representative test specimen from at least every tenth batch of each type of wet radioactive waste (e.g., filter sludges, spent resins, evaporator bottoms, boric acid solutions, and sodium sulfate solutions).

a. If any test specimen fails to verify SOLIDIFICATION, the SOLIDIFICATION of the batch under test shall be suspended until such time as additional test specimens can be obtained, alternative SOLIDIFICATION parameters can be determined in accordance with the PROCESS CONTROL PROGRAM, and a subsequent test verifies SOLIDIFICATION. SOLIDIFICATION of the batch may then be resumed using the alternative SOLIDIFICATION parameters determined by the PROCESS CONTROL PROGRAM.

b. If the initial test specimen from a batch of waste fails to verify SOLIDIFICATION, the PROCESS CONTROL PROGRAM shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least 3 consecutive initial test specimens demonstrate SOLIDIFICATION. The PROCESS CONTROL PROGRAM shall be modified as required, as provided in Specification 6.14, to assure SOLIDIFICATION of subsequent batches of waste.

Reports

The semiannual Radioactive Effluent Release Report shall include the following information for each type of solid waste shipped offsite during the report period:

a. Container volume,

b. Total curie quantity (determined by measurement or estimate),

c. Principal radionuclides (determined by measurement or estimate),.

d. Type of waste (e.g., spent resin, compacted dry waste evaporator bottoms),

e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f. Solidification agent (e.g., cement, urea formaldehyde).

4.25 (Continued)

Bases

This specification implements the requirements of 10 CFR Part 50.36a and General Design Criterion 60 of Appendix A to 10 CFR Part 50. The process parameters included in establishing the PROCESS CONTROL PROGRAM may include, but are not limited to waste type, waste pH, waste/liquid/solidification agent/catalyst ratios, waste oil content, waste principal chemical constituents, mixing and curing times.

4.26 RADIOLOGICAL ENVIRONMENTAL MONITORING

Surveillance Requirements

The radiological environmental monitoring samples shall be collected per Table 3.26-1 from the locations shown in the ODCM and shall be analyzed to the requirements of Tables 3.26-1 and 4.26-1.

Bases

The radiological monitoring program required by this specification provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures of individuals resulting from the station operation. This monitoring program thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and modeling of the environmental exposure pathways. The specified monitoring program is in effect at the present time. Program changes may be initiated based on operational experience and changes in regional population or agricultural practices. The sample locations have been listed in the ODCM to retain flexibility for making changes as needed.

Table 4.26-1

MAXIMUM VALUES FOR THE LOWER LIMITS OF DETECTION (LLD)[a, d]

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m^3)	Fish (pCi/gm, dry)	Milk (pCi/l)	Food Products (pCi/gm, dry)	Mud and Silt (pCi/gm, dry)
gross beta	4[b]	1×10^{-2}	1×10^{-1}		1×10^{-1}	2×10^{-1}
^{3}H	2000 (1000[b])					
^{54}Mn	15					
^{59}Fe	30					
$^{58,60}Co$	15					
^{65}Zn	30					
^{95}Zr-Nb	15					
^{131}I	1	7×10^{-2}		1		
^{134}Cs	15	1×10^{-2}[c]				
^{137}Cs	18	1×10^{-2}[c]				
^{140}Ba-La	15					

Table 4.26-1 (Continued)

Table Notation

a - The LLD is defined in the ODCM.

Analyses shall be performed in such a manner that the stated LLD's will be achieved under routine conditions. Occasionally, background fluctuations, unavoidably small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLD's unachievable. In such cases, the contributing factors will be identified and described in the Annual Radiological Environmental Operating Report.

b - LLD for drinking water.

c - LLD shown is for composite analysis. For individual samples, 5×10^{-2} pCi/m^3 is the LLD.

d - Other peaks which are measurable and identifiable, together with the nuclides in Table 4.26-1, shall be identified and reported.

4.27 LAND USE CENSUS

Surveillance Requirements

The land use census shall be conducted annually by using methods that will provide the best results, such as door-to-door survey, aerial survey, or by consulting local agriculture authorities.

Reports

The results of the land use census shall be included in the Annual Radiological Environmental Operating Report.

Bases

This specification is provided to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. Restricting the census to gardens of greater than 500 square feet provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored, since a garden of this size is the minimum required to produce the quantity (26 kg/year) of leafy vegetable assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were used: (1) that 20% of the garden was used for growing broad-leaf vegetation (i.e., similar to lettuce and cabbage), and (2) a vegetation yield of 2 kg/square meter.

4.28 EXPLOSIVE GAS MIXTURE

Surveillance Requirements

The concentration of oxygen in the waste gas hold-up system shall be determined to be within the limits specified in 3.28 by continuously monitoring the waste gases in the waste gas hold-up system with the oxygen monitor demonstrated OPERABLE according to table 4.28-1. If the continuous monitor is inoperable, a daily sample will be taken and analyzed; during heatup or cooldown, a sample will be taken and analyzed within four hours.

Bases

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the waste gas treatment system is maintained below the flammability limits of hydrogen and oxygen. Maintaining the concentration of oxygen below the flammability limit provides the assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

TABLE 4.28-1
EXPLOSIVE GAS MIXTURE INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

Instrument	Instrument Channel Check	Source Check	Instrument Channel Calibration	Channel Test
Waste Gas Holdup System Oxygen Monitor	D	NA	Q(1)	M

Table Notation

(1) The Channel Calibration shall include the use of standard gas samples containing:

a. Nominal zero volume percent oxygen, balance nitrogen.

b. Nominal four volume percent oxygen, balance nitrogen.

4.29 FUEL CYCLE DOSE

Surveillance Requirements

Cummulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 3.21.2.a, 3.21.2.b, 3.22.1.a, 3.22.1.b, 3.22.2.a, 3.22.2.b, 3.22.3.a, and 3.22.3.b, and in accordance with the Offsite Dose Calculation Manual (ODCM).

Reports

Special reports shall be submitted as required under Specification 3.29.

Bases

This specification is provided to meet the dose limitations of 40 CFR 190. The specification requires the preparation and submittal of a Special Report whenever the calculated doses from plant radioactive effluents exceed twice the design objective doses of Appendix I. For the Rancho Seco site, it is highly unlikely that the resultant dose to a member of the public will exceed the dose limits of 40 CFR 190 if the plant remains within the reporting requirement level. The Special Report will describe a course of action which should result in the limitation of the annual dose to a member of the public to within the 40 CFR 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the member of the public from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 5 miles must be considered. If the dose to any member of the public is estimated to exceed the requirements of 40 CFR 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR 190 have not already been corrected), in accordance with the provisions of 40 CFR 190.11, is considered to be a timely request and fulfills the requirements of 40 CFR 190 until NRC staff action is completed. An individual is not considered a member of the public during any period in which he/she is engaged in carrying out any operation which is part of the nuclear fuel cycle.

4.30 INTERLABORATORY COMPARISON PROGRAM SURVEILLANCE REQUIREMENT

Surveillance Requirement

A summary of the results obtained as a participant of the Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Operating Report.

Bases

The requirement for participation in an Interlaboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental samples are performed as part of the quality assurance program for environmental monitoring in order to demonstrate that the results are reasonably valid.

5. DESIGN FEATURES

5.1 SITE

Specification

The Rancho Seco reactor is located on the 2,480 acres owned by Sacramento Municipal Utility District, 26 miles north-northeast of Stockton and 25 miles southeast of the City of Sacramento, California. FSAR Figure 1.1-2 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 2,100 feet.[1], [2]

REFERENCES

(1) FSAR paragraph 1.2.1

(2) FSAR paragraph 2.2.1

(3) Map: Figure 3.22-1

5.2 CONTAINMENT

Specification

The containment for this unit consists of two systems, which are the Reactor Building and Reactor Building Isolation Systems.

5.2.1 Reactor Building

The Reactor Building completely encloses the reactor and the associated reactor coolant system. It is a reinforced concrete structure in the shape of a cylinder with a shallow domed roof and a flat foundation slab. The cylindrical portion is prestressed by a post tension system consisting of horizontal and vertical tendons. The dome has a 3-way post tension system. The structure can withstand the loss of any three horizontal and any three vertical tendons in the cylinder wall and any three tendons in the dome without loss of function. The foundation slab is conventionally reinforced concrete, with high-strength reinforcing steel. The entire structure is lined with 1/4-inch welded steel plate to provide vapor tightness.

The free internal volume of the Reactor Building is approximately 1.98 x 10^6 cubic feet. The approximate inside dimensions are: diameter, 130 feet; height, 185 feet. The approximate thicknesses of the concrete forming the buildings are: Cylindrical Wall 3-feet, 9-inches; Dome 3-feet, 6-inches; and the Foundation Slab 8-feet, 6-inches.

The concrete containment structure provides adequate biological shielding for both normal operation and accident situations. Design pressure and temperature are 59 psig and 286 F, respectively.

The Reactor Building is designed for an external atmospheric pressure of 2.0 psi greater than the internal pressure. This corresponds to the differential pressure that could be developed if the building is sealed with an internal temperature of 120 F with a barometric pressure of 29.0 inches of Hg and the building is subsequently cooled to an internal temperature of 80 F with concurrent rise in barometric pressure to 31.0 inches of Hg. Since the building is designed for this pressure differential, vacuum breakers are not required.

5.2.1 (Continued)

Penetration assemblies are structurally welded to the Reactor Building liner to form a seal. Access openings, electrical penetration cannister, and the fuel transfer tube covers are equipped with double seals. Reactor Building purge penetrations are equipped with double valves having resilient seating surfaces.[1]

The principal design basis for the structure is that it be capable of withstanding the internal pressure resulting from a loss of coolant accident, as defined in FSAR Section 14, with no loss of integrity. In this event, the total energy contained in the water of the reactor coolant system is assumed to be released into the Reactor Building through a break in the reactor coolant piping. Subsequent pressure behavior is determined by the building volume, safety features, and the combined influence of energy sources and heat sinks.

5.2.2 Reactor Building Isolation System

Leakage through all fluid penetrations not serving accident-consequence-limiting systems is to be minimized by a double barrier so that no single, credible failure or malfunction of an active component can result in loss-of-isolation or intolerable leakage. The installed double barriers take the form of closed piping systems, both inside and outside the Reactor Building, and various types of isolation valves.[2]

REFERENCES

(1) FSAR paragraph 5.2.3

(2) FSAR Section 5.2.4

5.3 REACTOR

Specification

5.3.1 Reactor Core

5.3.1.1 The reactor core contains approximately 93.1 metric tons of slightly enriched uranium dioxide pellets. The pellets are encapsulated in zircaloy-4 tubing to form fuel rods. The reactor core is made up of 177 fuel assemblies. Each fuel assembly contains 208 fuel rods.[1][2]

5.3.1.2 The reactor core shall approximate a right circular cylinder with an equivalent diameter of 128.9 inches and an active height of 144 inches.[2]

5.3.1.3 The average enrichment of the initial core for Rancho Seco is a nominal 2.57 weight percent of U^{235}. Three fuel enrichments are used in the initial core.

5.3.1.4 There are 61 full-length control rod assemblies (CRA) and 8 axial power shaping rod assemblies (APSRA) distributed in the reactor core as shown in FSAR Figure 3.2-45. The full-length CRA contains a 134-inch length of silver-indium-cadmium alloy clad with stainless steel. The APSRA contains a 36-inch length of silver-indium-cadmium alloy clad with stainless steel.[3]

5.3.1.5 The initial core will have 68 burnable poison as[illegible]es with similar dimensions as the full-length control r[illegible]he cladding will be zircaloy-4 filled with aluminum oxide-boron carbide pellets and placed in the core as shown in FSAR Figure 3.2-2.

5.3.1.6 Reload fuel assemblies and rods shall conform to design and evaluation described in the FSAR. A reload core may also have burnable poison assemblies with dimensions similar to the full-length control rods with materials as specified in 5.3.1.5.

5.3.2 Reactor Coolant System

5.3.2.1 The reactor coolant system shall be designed and constructed in accordance with code requirements.[4]

5.3.2.2 The reactor coolant system and any connected auxiliary systems exposed to the reactor coolant conditions of temperature and pressure, shall be designed for a pressure of 2,500 psig and temperature of 650 F. The pressurizer and pressurizer surge line shall be designed for a temperature of 670 F.[5]

5.3.2.3 The reactor coolant system volume shall be less than 12,200 cubic feet.

REFERENCES

(1) FSAR Table 3.2-1

(2) FSAR Table 3.2-2

(3) FSAR Paragraph 3.2.4.2

(4) FSAR Paragraph 4.1.3

(5) FSAR Paragraph 4.1.2

5.4 NEW AND SPENT FUEL STORAGE FACILITIES

Specification

5.4.1 New Fuel Inspection and Temporary Storage Rack

A. New fuel shall be removed from the shipping containers, inspected, and temporarily stored in the new fuel dry storage rack or stored in the pool. The dry storage rack is located on the operating floor and consists of two parallel modules containing ten spaces each on 21-1/8 inch centers. This spacing is sufficient to maintain K_{eff} less than 0.9 when flooded with unborated water, based on a fuel enrichment of 3.5 weight percent U^{235}. If the fuel assemblies have been stored in the dry storage rack, after inspection they may be moved to the new fuel elevator and lowered to the floor of the spent fuel storage pool, one at a time.(1)

B. New fuel may also be stored in their shipping containers.

5.4.2 New and Spent Fuel Storage Racks and Failed Fuel Storage Container Rack

New fuel, while awaiting transfer to the Reactor Building, and irradiated or failed fuel, prior to off-site shipment, will be stored in the stainless steel lined pool. The spent fuel pool is sized to accommodate 579 fuel assemblies, including three assemblies in failed fuel containers. During refueling, the borated fuel pool water will have a minimum concentration of 1800 ppm.

The pool has the capability of storing new and spent fuel assemblies in 14 stainless steel rack modules and three failed fuel assemblies in a special rack module. All assemblies are on minimum 14.5-inch centers in both directions. This spacing is sufficient to maintain K_{eff} less than 0.95 when flooded with unborated water, based on a fuel enrichment of 3.5 weight percent.

5.4.3 New and Spent Fuel Temporary Storage

The Reactor Building has one single-row, stainless steel storage rack in the deep portion of the refueling canal. This rack is designed to hold six assemblies and one failed fuel detection can, all on 21-1/8-inch centers.

5.4.4 Spent Fuel Pool and Storage Rack Design

The spent fuel pool and all storage racks are designed for the design basis earthquake.

REFERENCE

(1) FSAR Subsection 9.8

6. ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Manager of Nuclear Operations shall be responsible for the management of the overall facility, and the Plant Superintendent shall be responsible to him for the operation and maintenance of the plant. They shall delegate in writing the succession to their responsiblity during their absences.

6.2 ORGANIZATION

6.2.1 Offsite

The offsite organization for the facility management and technical support shall be as shown in Figure 6.2-1.

6.2.2 Facility Staff

The Facility organization shall be as shown in Figure 6.2-2 and;

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

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

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

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

e. ALL CORE ALTERATIONS, after the initial fuel loading, shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator limited to Fuel Handling who has no other concurrent responsibilities during this operation.

f. A site Fire Brigade of at least 5 members shall be maintained onsite at all times.* The Fire Brigade shall not include 3 members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency.

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* Fire Brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence provided immediate action is taken to fill the required positions.

TABLE 6.2-1

SHIFT CREW PERSONNEL AND LICENSE REQUIREMENTS

RANCHO SECO JOB TITLE	REACTOR MODE	
	COLD SHUTDOWN	OTHER THAN COLD SHUTDOWN
Shift Supervisor	1-SL	1-SL
Sr. Control Room Operator or Control Room Operator	1-L	2-L*
Auxiliary Operator or Equipment Attendant	1	1
Equipment Attendant or Power Plant Helper		1
Shift Technical Advisor	0	1
Minimum Total Personnel**	3	6

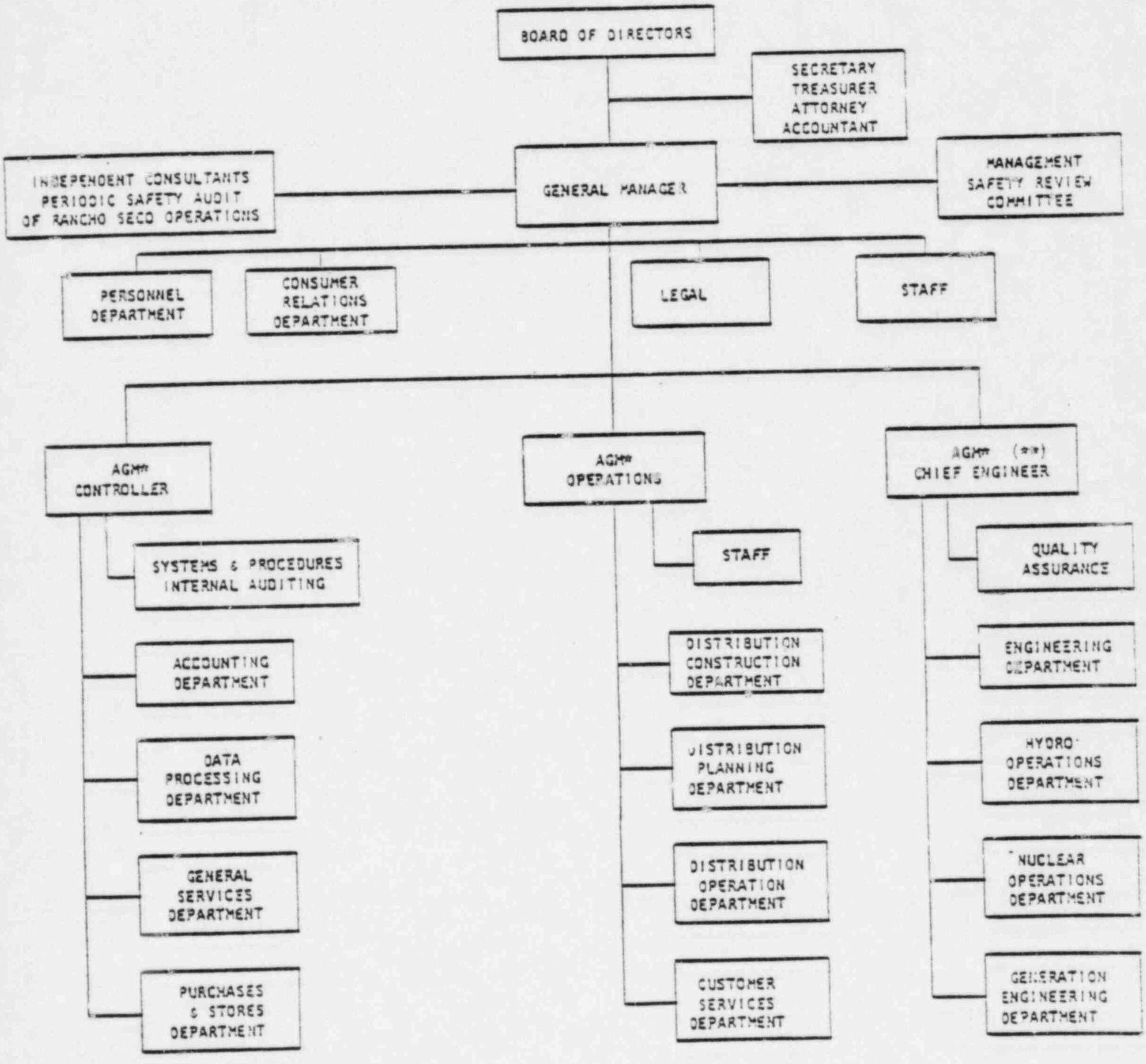
* One licensed operator when the reactor is shutdown greater than 1% k/k.

** In the event that any member of a minimum shift crew is absent or incapacitated due to illness or injury, a qualified replacement shall be designated to report onsite within two hours.

SL - NRC Senior Licensed Operator

L - NRC Licensed Operator

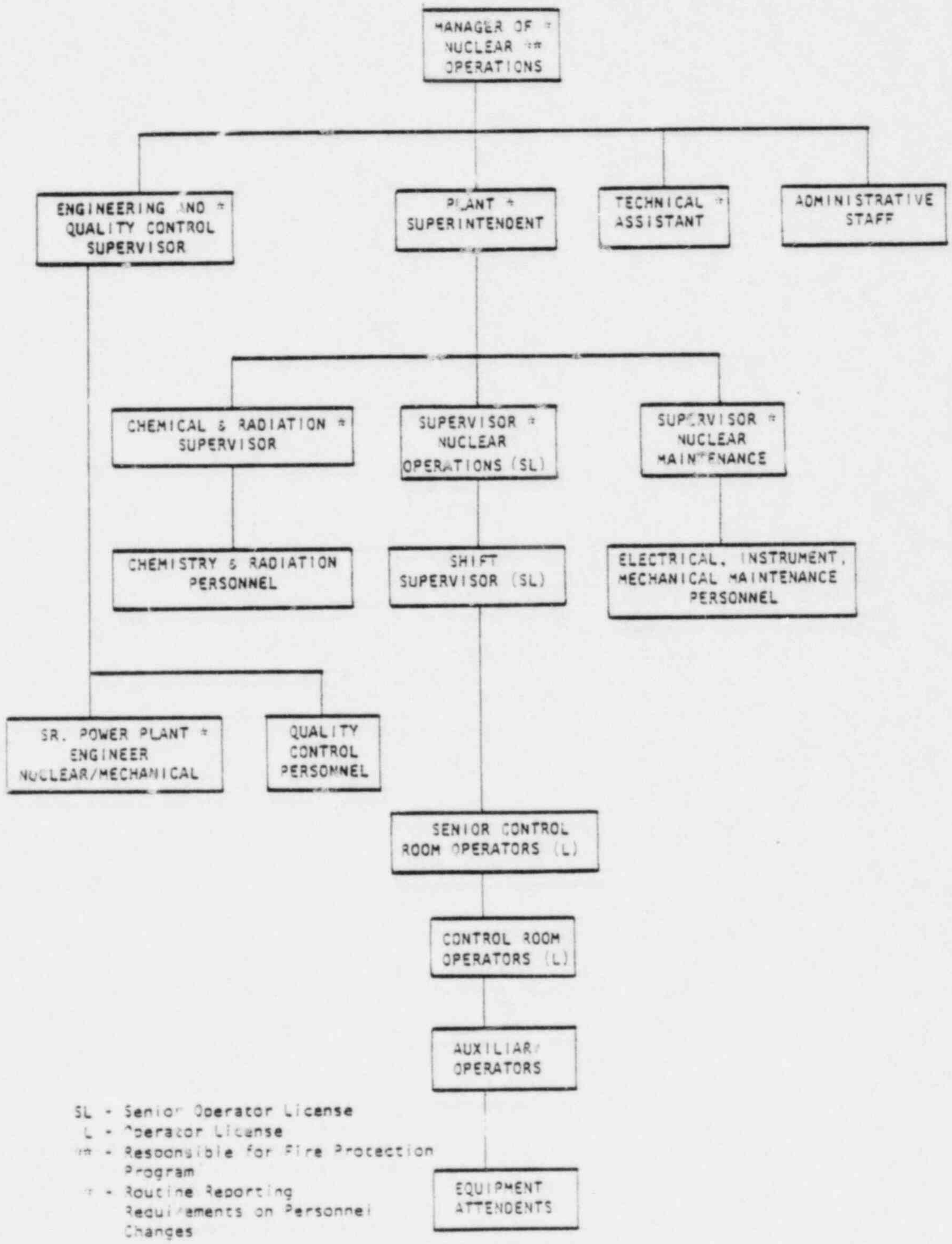
FIGURE 6.2-1
SMUD ORGANIZATION CHART



* AGM - Assistant General Manager
** Responsible for Fire Protection Program

Figure 6.2-2

PLANT ORGANIZATION CHART



6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the operating staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for (1) the Chemical-Radiation Supervisor who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975 and (2) the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline. The STA shall receive specific training in plant design, and response and analysis of the plant for transients and accidents.

6.4 TRAINING

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

6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the Safety Technician and shall meet or exceed the requirements of Section 27 of the NFPA Code - 1975, except, refresher classroom training shall be on a quarterly schedule.

6.5 REVIEW AND AUDIT

6.5.1 Plant Review Committee (PRC)

6.5.1.1 Function of Plant Review Committee

The Plant Review Committee shall function to advise the Manager of Nuclear Operations and the Plant Superintendent on all matters related to nuclear safety.

6.5.1.2 Composition of Plant Review Committee

The Plant Review Committee shall be composed of the:

Chairman:	Technical Assistant
Member:	Nuclear Operations Superintendent
Member:	Engineering & Quality Control Superintendent
Member:	Nuclear Maintenance Superintendent
Member:	Chemistry & Radiation Protection Superintendent

Other members as the Manager Of Nuclear Operations may appoint from time-to-time.

6.5.1.3 Alternates

Alternate Chairman or members shall be appointed, in writing, by the Manager of Nuclear Operations to serve on a temporary basis; however, no more than two alternates shall participate in PRC activities at any one time.

6.5.1.4 Meeting Frequency

The PRC shall meet a least once per calendar month and as convened by the PRC Chairman.

6.5.1.5 Quorum

A Quorum of PRC shall consist of the Chairman and two members, including alternates.

6.5.1.6 Responsibilities

The Plant Review Committee shall be responsible for:

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

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

6.5.1.6 (continued)

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

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

e. Investigation of all violations of the Technical Specifications, and shall prepare and forward a report covering evaluation and recommendations to prevent recurrence to the Plant Superintendent, Manager of Nuclear Operations, and to the Chairman of the Management Safety Review Committee.

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

g. Performance of special reviews and investigations and reports thereon as requested by the Chairman of the Management Safety Review Committee.

h. Review of the Plant Security Plan and implementing procedures, and shall submit recommended changes to the Plan to the Chairman of the Management Safety Review Committee.

i. Review of the Emergency Plan and implementing procedures and shall submit recommended changes to the Chairman of the Management Safety Review Committee.

j. Review of every unplanned onsite release of radioactive material to the environs; evaluate the event; specify remedial action to prevent recurrence; and document the event description, evaluation, and corrective action, and the disposition of the corrective action in the plant records.

k. Review of major changes to radwaste systems.

6.5.1.7 Authority

The Plant Review Committee shall:

a. Recommend to the Plant Superintendent written approval or disapproval of items considered under 6.5.1(a) through (d) above.

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

6.5.1.7 (continued)

c. Provide immediate written notification to the Chairman of the Management Safety Review Committee of disagreement between the PRC and the Manager of Nuclear Operations or the Plant Superintendent; however, the Manager of Nuclear Operations shall have responsibility for resolution of such disagreements pursuant to 6.5.1.1 above.

6.5.1.8 Records

The Plant Review Committee shall maintain written minutes of each meeting, and copies shall be provided to the Plant Superintendent, Manager of Nuclear Operations, and the Chairman of the Management Safety Review Committee.

6.5.2 Management Safety Review Committee

6.5.2.1 Function

The Management Safety Review Committee shall function to provide independent review and audit of designated activites in the areas of:

a. Nuclear Power Plant operations.

b. Nuclear engineering.

c. Chemistry and Radiochemistry.

d. Metallurgy.

e. Instrumentation and Control.

f. Radiological safety

g. Mechanical and electrical engineering.

h. Quality assurance practices.

6.5.2.2 Composition

The MSRC shall be composed of the:

Assistant General Manager & Chief Engineer - Chairman
Technical Assistant - Secretary
Manager, Engineering Department
Assistant General Manager, Operations
Manager, Nuclear Operations
Supervising Nuclear Engineer
Manager, Generation Engineering
Quality Assurance Director
Quality Assurance Engineer
Plant Superintendent

6.5.2.3 Alternates

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

6.5.2.4 Consultants

Consultants shall be utilized as determined by the MSRC Chairman to provide expert advice to the MSRC.

6.5.2.5 Meeting Frequency

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

6.5.2.6 Quorum

A quorum of MSRC shall consist of the Chairman or his designated alternate and a majority of the MSRC members, including alternates. No more than a minority of the quorum shall have line responsibility for operation of the facility.

6.5.2.7 Review

The MSRC shall review:

a. The safety evaluation for (1) changes to procedures, equipment, or systems, and (2) tests or experiements completed under the provision of Section 50.59.10 CFR, to verify that such actions did not constitute an unreviewed safety question.

6.5.2.7 (continued)

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

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

d. Proposed changes in Technical Specifications or licenses.

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

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

g. Events requiring 24-hour written notification to the Commission.

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

i. Reports and meeting minutes of the Plant Review Committee.

6.5.2.8 Audits

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

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

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

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

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

6.5.2.8 (continued)

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

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

g. Any other area of facility operation considered appropriate by the MSRC or the General Manager.

h. Compliance with fire protection requirements and implementing procedures at least once per two years.

i. An independent fire protection and loss prevention inspection and audit shall be performed annually, utilizing either qualified offsite licensee personnel or an outside fire protection firm.

j. An inspection and audit of the fire protection and loss prevention program shall be performed by an outside qualified fire consultant at intervals no greater than three years.

k. The radiological environmental monitoring program and the results thereof at least once per 12 months.

l The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once per 24 months.

m. The PROCESS CONTROL PROGRAM for the SOLIDIFICATION of radioactive wastes from liquid systems at least once per 24 months.

6.5.2.9 Authority

The MSRC shall report to and advise the General Manager on those areas of responsibility specified in Sections 6.5.2.7 and 6.5.28.

6.5.2.10 Records

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

a. Minutes of each MSRC meeting shall be prepared, approved, and forwarded to the General Manager within 14 days following each meeting.

6.5.2.10 (continued)

b. Reports of reviews encompassed by Section 6.5.2.7(e), (f), (g), and (h), above, shall be prepared, approved, and forwarded to the General Manager within 14 days following completion of the review.

c. Audit reports encompassed by Section 6.5.2.8, above, shall be forwarded to the General Manager and the management positions responsible for the areas audited within 30 days after completion of the audit.

6.6 REPORTABLE OCCURRENCE ACTION

6.6.1 The following actions shall be taken in the event of a REPORTABLE OCCURRENCE:

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

b. Each Reportable Occurrence shall be reviewed by the PRC and the report shall be submitted to the MSRC, Manager of Nuclear Operations and the Plant Superintendent prior to its submittal to the Commission.

6.7 SAFETY LIMIT VIOLATION

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

a. The unit shall be placed in at least Hot Shutdown WITHIN ONE HOUR.

b. The Safety Limit Violation shall be reported to the Plant Superintendent, the Manager of Nuclear Operations, the Chairman of the MSRC, and to the Commission within 24 hours,

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

d. The Safety Limit Violation Report shall be submitted to the Commission, the MSRC, the Manager of Nuclear Operations, and the Plant Superintendent within 14 days of the violation.

6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, November 1972.

b. Refueling operations.

c. Surveillance and test activities of safety-related equipment.

d. Security Plan implementation.

e Emergency Plan implementation.

f. Process control program implementation.

g. Offsite Dose Calculation Manual implementation.

h. Effluent and environmental quality control program.

6.8.2 Each procedure and administrative policy of 6.8.1, above, and changes thereto, shall be reviewed by the PRC. Those matters pertaining to items 6.8.1a, b, c, f, and g, above, shall be approved by the Plant Superintendent prior to implementation and reviewed periodically as set forth in each document. The manager of Nuclear Operations shall approve Security Plan and Emergency Plan implementing procedures.

6.8.3 Temporary changes to procedures 6.8.1, above, may be made, provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed by the PRC, and approved by the Plant Superintendent within seven days of implementation.

6.9 REPORTING REQUIREMENTS

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Director of the Regional Office of Inspection and Enforcement, unless otherwise noted.

6.9.1. Annual Reports

Annual reports covering the activities of the unit, as described below, for the previous calendar year shall be submitted prior to March 1 of each year following initial criticality

Reports required on an annual basis shall include:

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure, according to work and job functions,[(2)] e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures, totaling less than 20% of the individual total dose, need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

6.9.2 Annual Radiological Environmental Operating Report

6.9.2.1 Routine radiological environmental operating reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following initial criticality.

6.9.2.2 The annual radiological environmental operating reports shall include summaries, interpretations, and statistical evaluation of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, operational controls (as appropriate), and previous environmental surveillance reports, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of the land use censuses. If harmful effects or evidence of irreversible damage are detected by the monitoring, the report shall provide an analysis of the problem and a planned course of action to alleviate the problem.

6.9.2.2 (continued)

The annual radiological environmental operating reports shall include summarized and tabulated results in the format of Table 6.9-1, of all radiological environmental samples taken during the report period. In the event that some results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; including sampling methods for each sample type, size and physical characteristics of each sample type, sample preparation methods, analytical methods, and measuring equipment used; a map of all sampling locations keyed to a table giving distances and directions from one reactor; the result of land use censuses, and the results of licensee participation in the Interlab Comparison Program. The annual report shall also include information related to Specification 4.2.9.

6.9.3 Semi-Annual Radioactive Effluent Release Report

Routine radioactive effluent release reports covering the operation of the unit during the previous six months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

6.9.3.1 The radioactive effluent release reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," with data summarized on a quarterly basis, following the format of Appendix B thereof.

The radioactive effluent release reports shall include the release of gaseous effluents during each quarter, as outlined in Regulatory Guide 1.21, with the data summarized on a quarterly basis, following the format of Appendix B thereof. A summary of meteorological conditions during the release of gaseous effluents will be retained on-site for two years. In addition, any changes to the Offsite Dose Calculation Manual will be submitted with the Semiannual Radioactive Effluent Release Report.

6.9.3.1(Continued)

The radioactive effluent release reports shall include an assessment of the radiation doses from radioactive effluents to individuals due to their activities inside the site boundary during the report period. All assumptions used in making these assessments (e.g., specific activity, exposure time, and location) shall be included in these reports.

The radioactive effluent release reports shall include the following information for all unplanned releases to unrestricted areas of radioactive materials in gaseous and liquid effluents:

a. A description of the event and equipment involved.

b. Cause(s) for the unplanned release.

c. Actions taken to prevent recurrence.

d. Consequences of the unplanned release.

The radioactive effluent release reports shall include an assessment of radiation doses from the radioactive liquid and gaseous effluents released from the unit during each calendar quarter, as outlined in Regulatory Guide 1.21. In addition, the nearest offsite receptor maximum noble gas gamma air and beta air dose shall be evaluated. The releases of effluents shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the Offsite Dose Calculation Manual (ODCM).

The radioactive effluent release reports shall include any changes to the PROCESS CONTROL PROGRAM (PCP) or (ODCM) made during the reporting period, as provided in Specifications 6.14 and 6.15.

6.9.4 Monthly Report

Routine reports of operating statistics, including narrative summary of operating and shutdown experience, or major safety-related maintenance, and tabulations of facility changes (including changes to radwaste treatment system), tests or experiments required pursuant to 10 CFR 50.59(b), shall be submitted on a monthly basis to the Office of Management Information and Program Control, U. S. Nuclear Regulatory Commission, Washington, D. C., 20555, with a copy to the Regional Office, postmarked not later than the 15th day of each month following the calendar month covered by the report.

6.9.5 Reportable Occurrences

The REPORTABLE OCCURRENCES of Specifications 6.9.5.1 and 6.9.5.2 below, including corrective actions and measures to prevent recurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a license event report shall be completed and reference shall be made to the original report date.

6.9.5.1 Prompt Notification with Written Follow-up

The types of events listed below shall be reported within 24 hours by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the Director of the Regional Office, or his designate, no later than the first working day following the event, with a written follow-up report within two weeks. The written follow-up report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Failure of the reactor protection system or other sytems subject to limiting safety system settings to initiate the required protective function by the time a monitored parameter reaches the setpoint specified as the limiting condition for operation established in the technical specifications.

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

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

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

6.9.5.1 (Continued)

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

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

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

h. Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the Safety Analysis Report, or in the bases for the Technical Specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

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

j. Offsite releases of radioactive materials in liquid and gaseous effluents in excess of the limits specified in sections 3.21.1 and 3.22.1.

k. Occurrence of radioactive material contained in gaseous and liquid holdup tanks in excess of that permitted by the limiting condition for operation established in the Technical Specifications.

6.9.5.2 Thirty Day Written Reports

The types of events listed below shall be the subject of written reports to the Director of the Regional Office within 30 days of occurrence of the event. The written report shall include, as a minimum, a completed copy of a Licensee Event Report form. Information provided on the Licensee Event Report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

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

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

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

d. Abnormal degradation of systems other than those specified in 6.9.4.1.c, above, designed to contain radioactive material resulting from the fission process.

e. An unplanned offsite release of: (1) more than one curie of radioactive material in liquid effluents, (2) more than 150 curies of noble gas in gaseous effluents, or (3) more than 0.05 curies of radioiodine in gaseous effluents. The report of an unplanned offsite release of radioactive material shall include the following information:

   1. A description of the event and equipment involved.

   2. Cause(s) for the unplanned release.

   3. Actions taken to prevent recurrence.

   4. Consequences of the unplanned release.

6.9.6 Special Reports

Special reports shall be submitted to the Director of the Regulatory Operations Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. A Reactor Building structural integrity report shall be submitted within 90 days of completion of each of the following tests covered by Technical Specification 4.4.2 (the integrated leak rate test is covered in Technical Specification 4.4.1.1):

   1. Annual Inspection (if corrective action necessary).

   2. Tendon Stress Surveillance.

   3. End Anchorage Concrete Surveillance.

   4. Liner Plate Surveillance.

b. Inservice Inspection Program

c. Reserved for Proposed Amendment No. 43.

d.	Status of Inoperable Fire Protection Equipment.	30 days	(3.14)
e.	Radioactive Liquid Effluent Concentration	14 days	(3.21.2)
f.	Radioactive Liquid Effluent Dose	30 days	(3.21.2)
g.	Gaseous Effluents	14 days	3.21.2)
h.	Noble Gas Limits	30 days	(3.22.2)
i.	Radioiodine and Particulates	30 days	(3.22.3)
j.	Gaseous Radwaste Treatment	30 days	(3.23)
k.	Gas Storage Tanks	14 days	(3.24)
l.	Radiological Monitoring Program	30 days	(3.26)
m.	Monitoring Point Substitutions	30 days	(3.26)
n.	Land Use Census	30 days	(3.27)
o.	Fuel Cycle Dose	60 days	(4.29)
p.	Liquid Holdup Tanks	30 days	(3.21.3)

6.10 RECORD RETENTION

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

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

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

c. REPORTABLE OCCURENCE Reports.

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

e. Records of reactor tests and experiments.

f. Records of changes made to Operating Procedures.

g. Records of radioactive shipments.

h. Records of sealed source leak tests and results.

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

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

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

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

c. Records of facility radiation and contamination surveys.

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

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

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

6.10.2 (Continued)

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

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

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

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

k. Records of meetings of PRC and MSRC.

l. Records for Environmental Qualification which are covered under provisions of paragraph 6.14.

m. Records for Environmental Monitoring Program.

6.11 RADIATION PROTECTION PROGRAM

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

6.12 Deleted.

6.13 HIGH RADIATION AREA

6.13.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c) (2) of 10 CFR 20:

a. Each High Radiation Area in which the intensity of radiation is greater than 100 mrem/hr but less than 1,000 mrem/hr shall be barricaded and conspicuously posted as a High Radiation Area and entrance thereto shall be controlled by issuance of a Radiation Work Permit, and any individual or group of individuals permitted to enter such areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. Each High Radiation Area in which the intensity of radiation is greater than 1,000 mrem/hr shall be subject to the provisions of 6.13.1(a) above, and, in addition, locked doors shall be provided to prevent unauthorized entry into such area, and the keys shall be maintained under the administrative control of the Shift Supervisor on duty. Certain areas within the Reactor Building may use conspicuous visible or audible signals such that an individual is made aware of the presence of the High Radiation Area, in lieu of locked doors.

6.14 PROCESS CONTROL PROGRAM (PCP)

6.14.1 Function

The PCP shall be a manual containing the equipment operating procedures, process parameters, set points, drawings and controls, and the laboratory procedures detailing the program of sampling, analysis, and evaluation within which solidification of radioactive wastes from liquid systems is assured, and the surveillance requirements of these Technical Specifications.

6.14.2 Changes to the PCP shall be made by either of the following methods:

A. Licensee initiated changes:

1. Shall be submitted to the Commission by inclusion in the semiannual Radioactive Effluent Release Report for the period in which the change(s) was/were made and shall contain:

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

   b. A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

   c. Documentation of the fact that the change has been reviewed and found acceptable by the Plant Review Committee.

2. Shall become effective upon review and acceptance by the PRC, unless otherwise acted upon by the Commission through written notification to the Licensee.

6.15 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.15.1 Function

The ODCM shall describe the methodology and parameters to be used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and in the calculation of gaseous and liquid effluent monitoring instrumentation alarm/trip setpoints consistent with the applicable LCO's contained in these Technical Specifications. Methodologies and calculational procedures acceptable to the Commission are contained in various Regulatory Guides as noted in the bases of applicable LCO's.

6.15.2 Any changes to the ODCM shall be made by either of the following methods:

A. Licensee-initiated changes:

1. Shall be submitted to the Commission by inclusion in the Semiannual Radioactive Effluent Release Report and shall contain:

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change;

   b. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

   c. Documentation of the fact that the change has been reviewed and found acceptable by both the PRC and MSRC.

2. Shall become effective upon a date specified and agreed to by both the PRC and MSRC following their review and acceptance of the change.

6.16 MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS (LIQUID, GASEOUS, AND SOLID)

6.16.1 Function

The radioactive waste treatment system (liquid, gaseous, and solid) are those systems described in the facility Final Safety Analysis Report or Hazards Summary Report, and amendments thereto, which are used to maintain that control over radioactive materials in gaseous and liquid effluents and in solid waste packaged for offsite shipment required to meet the LCO's set forth in these Specifications.

6.16.2 Major changes to the radioactive waste systems (liquid, gaseous, and solid) shall be made by either of the following methods. For the purpose of this specification, "major changes" is defined in Specification 6.16.3, below.

A. Licensee-initiated changes:

1. The Commission shall be informed of all changes by the inclusion of a suitable discussion of each change in the Annual FSAR Update for the period in which the changes were made. The discussion of each change shall contain:

   a. A summary of the evaluation that led to the determination that the change could be made (in accordance with 10 CFR 50.59);

   b. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

   c. A detailed description of the equipment, components, and processes involved, and the interfaces with other plant systems;

   d. An evaluation of the change which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste from those previously predicted in the license application and amendments thereto;

   e. An evaluation of the change which shows the expected maximum exposures to individuals at or beyond the SITE BOUNDARY and to the general population from those previously estimated in the license application and amendments thereto;

6.16.2 (Continued)

f. A comparison of the predicted releases of radioactive materials in liquid and gaseous effluents and in solid waste to the actual releases for the period in which the changes were made;

g. An estimate of the exposure to plant operating personnel as a result of the change; and

h. Documentation of the fact that the change was reviewed and found acceptable by both the PRC and MSRC.

2. The Change shall become effective upon review and acceptance by both the PRC and MSRC.

6.16.3 Background and definition of what constitutes "major changes" to radioactive waste systems (liquid, gaseous, and solid).

A. Background

1. 10 CFR Part 50, Section 50.34a(a) requires that each application to construct a nuclear power reactor provide a description of the equipment installed to maintain control over radioactive material in gaseous and liquid effluents produced during normal reactor operations, including operational occurrences.

2. 10 CFR Part 50, Section 50.34a (b) (2) requires that each application to construct a nuclear power reactor provide an estimate of the quantity of radionuclides expected to be released annually to unrestricted areas in liquid and gaseous effluents produced during normal reactor operation.

3. 10 CFR Part 50, Section 50.34a(3) requires that each application to construct a nuclear power reactor provide a description of the provisions for packaging, storage, and shipment offsite of solid waste containing radioactive materials resulting from treatment of gaseous and liquid effluents and from other sources.

4. 10 CFR Part 50, Section 50.34a(3)(c) requires that each application to operate a nuclear power reactor shall include (1) a description of the equipment and procedures for the control of gaseous and liquid effluents and for the maintenance and use of equipment installed in radioactive waste systems, and (2) a revised estimate of the information required in (b)(2) if the expected releases and exposures differ significantly from the estimate submitted in the construction permit.

6.16.3 (Continued)

5. The Regulatory staff's Safety Evaluation Report and Amendments thereto issued prior to the issuance of an operating license contains a description of the radioactive waste systems installed in the nuclear power reactor and a detailed evaluation (including estimated releases of radioactive materials in liquid and gaseous waste and quantities of solid waste produced from normal operation, estimated annual maximum exposures to an individual in the unrestricted area and estimated exposures to the general population) which shows the capability of these systems to meet the appropriate regulations.

6. The Regulatory staff's Final Environmental Statement issued prior to the issuance of an operating license contains a detailed evaluation as to the expected evnironmental impact from the estimated releases of radioactive material in liquid and gaseous effluents.

B. Definition

"Major Changes" to radioactive waste systems (liquid, gaseous, and solid) shall include the following:

1. Major changes in process equipment, components, structures, and effluent monitoring instrumentation from those described in the Final Safety Analysis Report (FSAR) or the Hazards Summary Report and evaluated in the staff's Safety Evaluation Report (SER) (e.g., deletion of evaporators and installation of demineralizers; use of fluidized bed calciner/incineration in place of cement solidification systems);

2. Major changes in the design of radwaste treatment systems (liquid, gaseous, and solid) that could significantly alter the characteristics and/or quantities of effluents released or volumes of solid waste stored or shipped offsite from those previously considered in the FSAR and SER (e.g., use of asphalt system in place of cement);

3. Changes in system design which may invalidate the accident analysis as described in the SER (e.g., changes in tank capacity that would alter the curies released); and

4. Changes in system design that could potentially result in a significant increase in occupational exposure of operating personnel (e.g., use of skid-mounted equipment, use of mobile processing equipment).

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