ARKANSAS NUCLEAR ONE
UNIT 1

LICENSE NO. DPR - 51

TECHNICAL SPECIFICATIONS

MAY 2 1 1974

ARKANSAS NUCLEAR ONE
UNIT 1

LICENSE NO. DPR-51

APPENDIX A
TECHNICAL SPECIFICATIONS

TABLE OF CONTENTS

INTRODUCTION

These Technical Specifications apply to Arkansas Nuclear One, 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.

LIST OF FIGURES

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 2568 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 the hot shutdown condition when it is subcritical by at least 1 percent $\Delta k/k$ 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 channels is less than 2 percent of 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 $\Delta k/k$ and the coolant temperature at the decay heat removal pump suction is at the refueling temperature (normally 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 Period

Time between normal refuelings of the reactor, not to exceed 24 months without prior approval of the AEC. As used in these technical specifications the refueling period refers to normal refueling.

1.2.9 Startup

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

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 and 7-9 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 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 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 set of two-out-of-four logic contacts from the four reactor protection channels.

1.4.5 Safety Features System

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 7-6 of the FSAR. The digital 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 weighted primary and secondary heat balance considering all heat losses. Between 0 and 15% power, only the primary heat balance is considered. From 15 to 100% power the heat balance is weighted linearly with only the secondary heat balance being considered at 100% power.

1.6 POWER DISTRIBUTION

1.6.1 Quadrant Power Tilt

Quadrant power tilt is defined by the following equation and is expressed in percent.

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

The power in any quadrant is determined from the power range channel displayed on the console for that quadrant. The average power is determined from an average of the outputs of the power range channels. If one of the power range channels is out of service, the remaining three operable power range channels or the incore detectors will be used to determine the average power. The quadrant power tilt limits as a function of power are stated in Specification 3.5.24.

1.6.2 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 expressed as a percentage of rated power. Imbalance is monitored continuously by the RPS using input from the power range channels. Imbalance limits are defined in Specification 2.1 and imbalance setpoints are defined in Specification 2.3.

1.7 REACTOR BUILDING

Reactor building integrity exists when the following conditions are satisfied:

a. The equipment hatch is closed and sealed and both doors of the personnel lock and emergency lock are closed and sealed, or b. below.

b. At least one door on each of the personnel lock and emergency lock is closed and sealed during personnel access or repair.

c. All non-automatic reactor building isolation valves and blind flanges are closed as required.

d. All automatic reactor building isolation valves are operable or deactivated in the closed position.

e. The reactor building leakage determined at the last testing interval satisfies Specification 4.4.1

1.8 ABNORMAL OCCURRENCE

An abnormal occurrence means the occurrence of any plant condition that results in:

1.8.1 A safety system setting less conservative than the limiting setting established in the Technical Specifications.

1.8.2 Violation of a limiting condition for operation established in the Technical Specifications.

1.8.3 An uncontrolled or unplanned release of radioactive material from any plant system designed to act as a boundary for such material in an amount of significance with respect to limits prescribed in Technical Specifications.

1.8.4 Failure of one or more components of an engineered safety feature or plant protection system that causes or threatens to cause the feature or system to be incapable of performing its intended function.

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

1.8.6 Uncontrolled or unanticipated changes in reactivity greater than 1% $\Delta k/k$.

1.8.7 Observed inadequacies in the implementation of administrative or procedural controls such that the inadequacy causes or threatens to cause the existence or development of an unsafe condition in connection with the operation of the plant.

1.8.8 Conditions arising from natural or man-made events that affect or threaten to affect the safe operation of the plant.

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 below and to the right of the pressure/temperature line 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 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

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 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 W-3 correlation.(1) The W-3 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 1.3. A

DNBR of 1.3 corresponds to a 94.3 percent probability at a 99 percent confidence level that DNB will not occur; this is considered a conservative margin to DNB for all operating conditions. 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 is nominally 45 psi; however, only a 30 psi drop was assumed in reducing the pressure trip set points 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 1.3 is predicted for the maximum possible thermal power (112 percent) when the reactor coolant flow is 131.3 × 10^6 lbs/h, which is the design flow rate for four operating reactor coolant pumps. This curve is based on the following nuclear power peaking factors(2) with potential fuel densification effects;

$$F_q^N = 2.67; \quad F_{\Delta H}^N = 1.78; \quad F_z^N = 1.50$$

These design limit power peaking factors are the most restrictive calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion, and form the core DNBR design basis.

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:

1. The 1.3 DNBR limit produced by a nuclear power peaking factor of F_q^N = 2.67 or the combination of the radial peak, axial peak and position of the axial peak that yields no less than 1.3 DNBR.

2. The combination of radial and axial peak that prevents central fuel melting at the hot spot. The limit is 20.1 kW/ft.

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

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 number of reactor coolant pumps in operation or the local quality at the point of minimum DNBR is equal to 15 percent(3), whichever condition is more restrictive.

Using a local quality limit of 15 percent at the point of minimum DNBR as a basis for curve 3 of Figure 2.1-3 is a conservative criterion even though the quality at the exit is higher than the quality at the point of minimum DNBR.

The DNBR as calculated by the W-3 correlation continually increases from the point of minimum DNBR, so that the exit DNBR is always higher and is a function of the pressure. Extrapolation of the W-3 correlation beyond its published quality range of +15 percent is justified on the basis of experimental data.(4)

The maximum thermal power for three pump operation is 86 percent due to a power level trip produced by the flux-flow ratio (75 percent flow × 1.07 = 80 percent power) plus the maximum calibration and instrumentation error. The maximum thermal power for other reactor coolant pump conditions is produced in a similar manner.

For each curve of Figure 2.1-3, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.3 or a local quality at the point of minimum DNBR less than 15 percent for that particular reactor coolant pump situation. The 1.3 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.

REFERENCES

(1) FSAR, Section 3.2.3.1.1

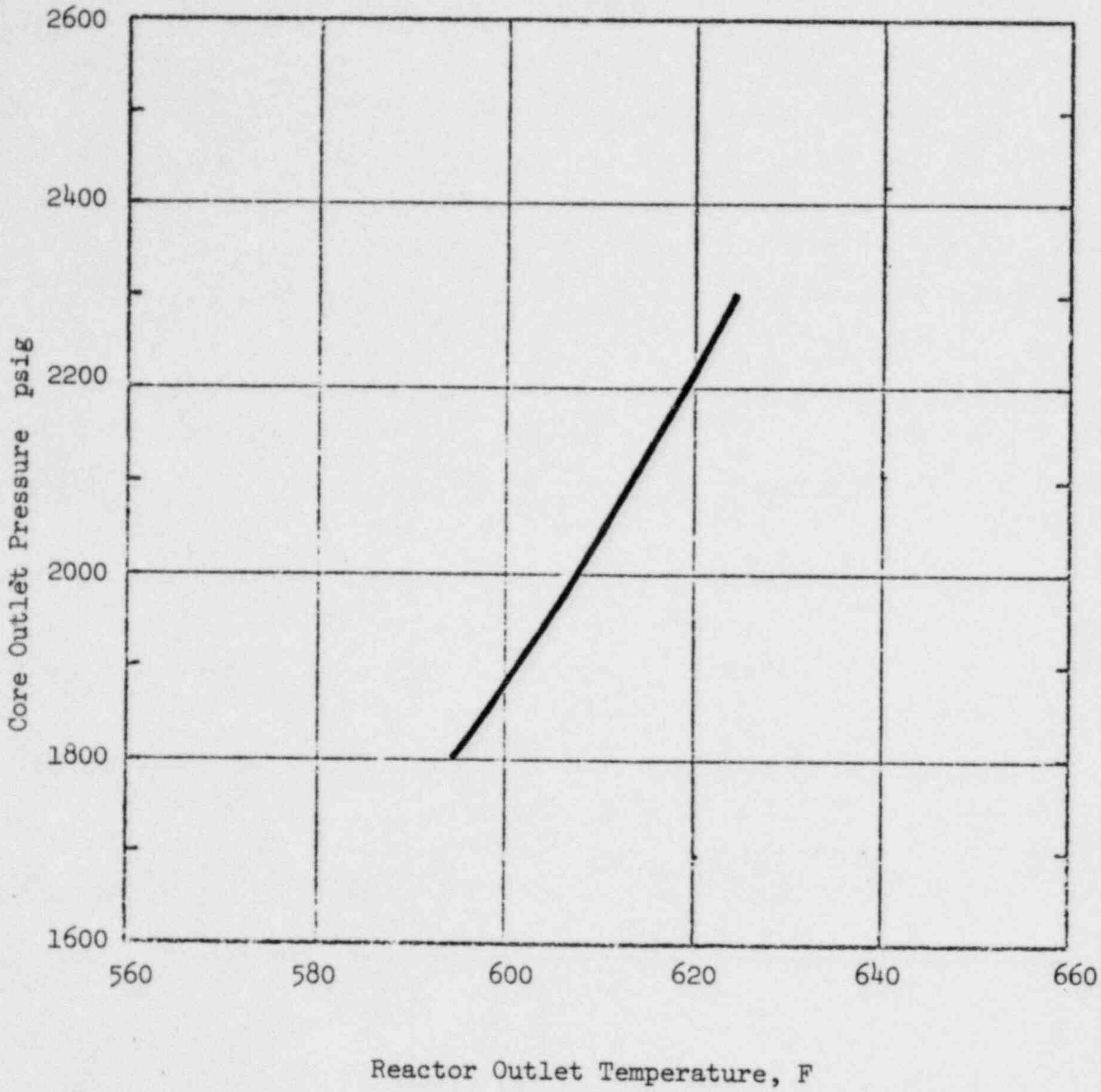
(2) FSAR, Section 3.2.3.1.1.c

(3) FSAR, Section 3.2.3.1.1.k

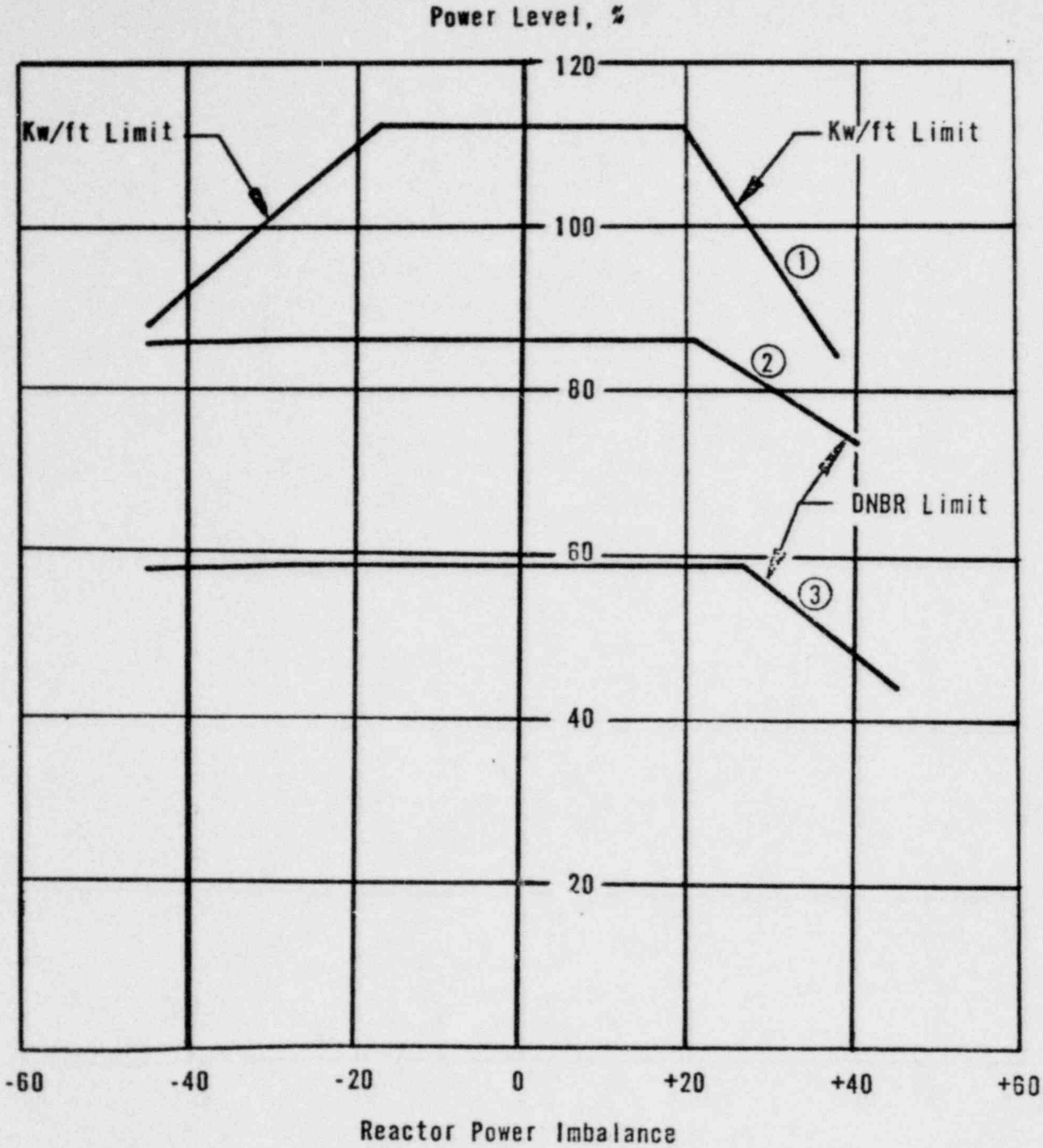
(4) The following papers which were presented at the Winter Annual Meeting, ASME, November 18, 1969, during the Two-Phase Flow and Heat Transfer in Rod Bundles Symposium:

(a) Wilson, et. al. "Critical Heat Flux in Non-Uniform Heater Rod Bundles."

(b) Gellerstedt, et. al. "Correlation of a Critical Heat Flux in a Bundle Cooled by Pressurized Water."

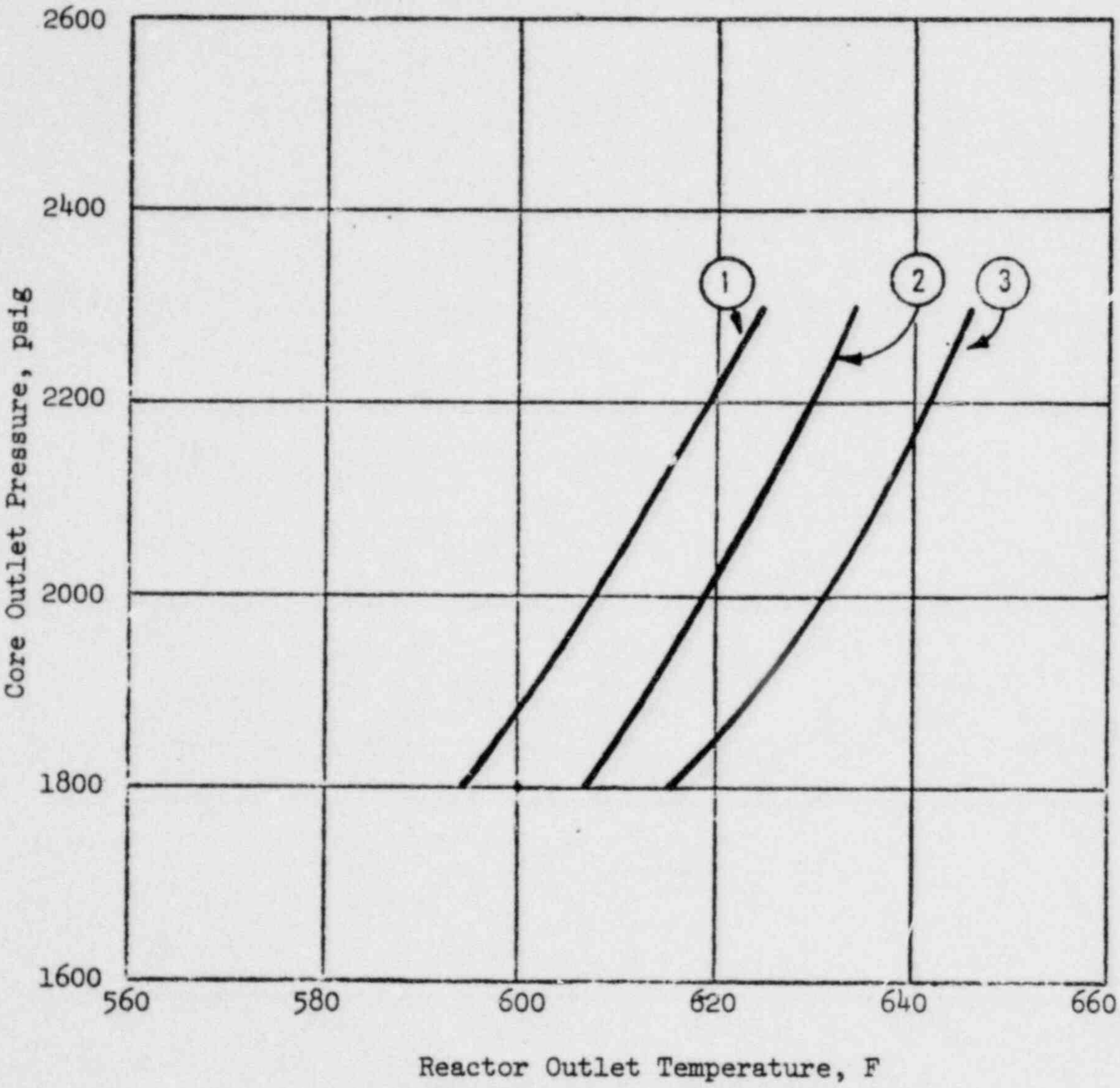


ARKANSAS POWER & LIGHT CO. ARKANSAS NUCLEAR ONE-UNIT 1	CORE PROTECTION SAFETY LIMIT	FIG. NO. 2.1-1



CURVE	REACTOR COOLANT FLOW (LB/HR)
1	131.3×10^6
2	98.1×10^6
3	64.4×10^6

ARKANSAS POWER & LIGHT CO. ARKANSAS NUCLEAR ONE-UNIT 1	CORE PROTECTION SAFETY LIMITS	FIG.NO. 2.1-2



CURVE	REACTOR COOLANT FLOW (LBS/HR)	POWER	PUMPS OPERATING (TYPE OF LIMIT)
1	131.3×10^6 (100%)	112	FOUR PUMPS (DNBR LIMIT)
2	98.1×10^6 (74.7%)	86	THREE PUMPS (DNBR LIMIT)
3	64.4×10^6 (49.0%)	58	ONE PUMP IN EACH LOOP(QUALITY LIMIT)

ARKANSAS POWER & LIGHT CO. ARKANSAS NUCLEAR ONE-UNIT 1	CORE PROTECTION SAFETY LIMITS	FIG. NO. 2.1-3

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 setpoint of the pressurizer code safety valves shall be in accordance with ASME, Boiler and Pressurizer Vessel Code, Section III, Article 9, Summer 1968.

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 pressure 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 (2355 psig) and the pressurizer code safety valves (2500 psig ±1%)(3) have been established to assure that the reactor coolant system pressure safety limit is not exceeded. The initial hydrostatic test is 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 2255 psig.(4)

REFERENCES

(1) FSAR, Section 4

(2) FSAR, Section 4.3.10.1

(3) FSAR, Section 4.2.4

(4) FSAR, Table 4-1

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 105.5 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%, which is more conservative than the value used in the safety analysis.

A. Overpower trip based on flow and imbalance

The power level trip set point produced by the reactor 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. Analysis has demonstrated that the specified power to flow ratio is adequate to prevent a DNBR of less than 1.3 should a low flow condition exist due to any electrical malfunction.

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 107 percent and reactor flow rate is 100 percent or flow rate is 93 percent and power level is 100 percent.

2. Trip would occur when three reactor coolant pumps are operating if power is 80 percent and reactor flow rate is 74.7 percent or flow rate is 70 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 52 percent and reactor flow rate is 49 percent or flow rate is 46 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 associated reactor power-to-reactor power imbalance boundaries by 1.07 percent for a 1 percent flow reduction.

B. Pump monitors

In conjunction with the power/imbalance/flow trip, the pump monitors prevent the minimum core DNBR from decreasing below 1.3 by tripping the reactor due to the loss of reactor coolant pump(s). The pump monitors also restrict the power level for the number of pumps in operation.

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 (2355 psig) has been established to maintain the system pressure below the safety limit (2750 psig) for any design transient.(2)

The low pressure (1800 psig) and variable low pressure ($16.25T_{out}$ -7756) trip setpoint 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 ($16.25T_{out}$ -7796).

D. Coolant outlet temperature

The high reactor coolant outlet temperature trip setting limit (619 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 reactor coolant system pressure trip.

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 administrative control the nuclear overpower trip set point must be reduced to a value ≤ 5.0 percent of rated power during reactor shutdown.

2. A high reactor coolant system pressure trip set point of 1720 psig is automatically imposed.

The purpose of the 1720 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 ≤ 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

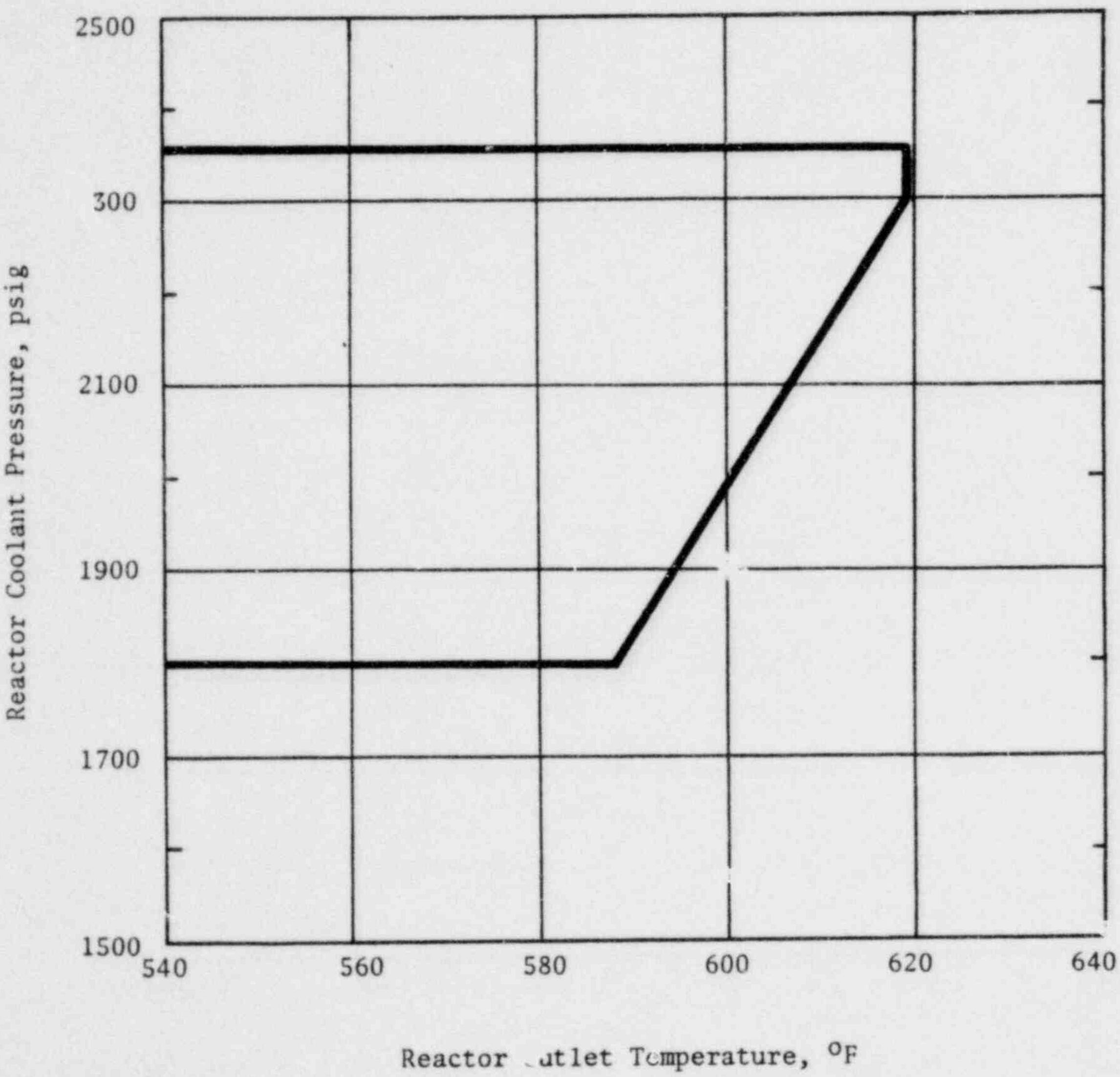
(1) FSAR, Section 14.1.2.3

(2) FSAR, Section 14.1.2.2

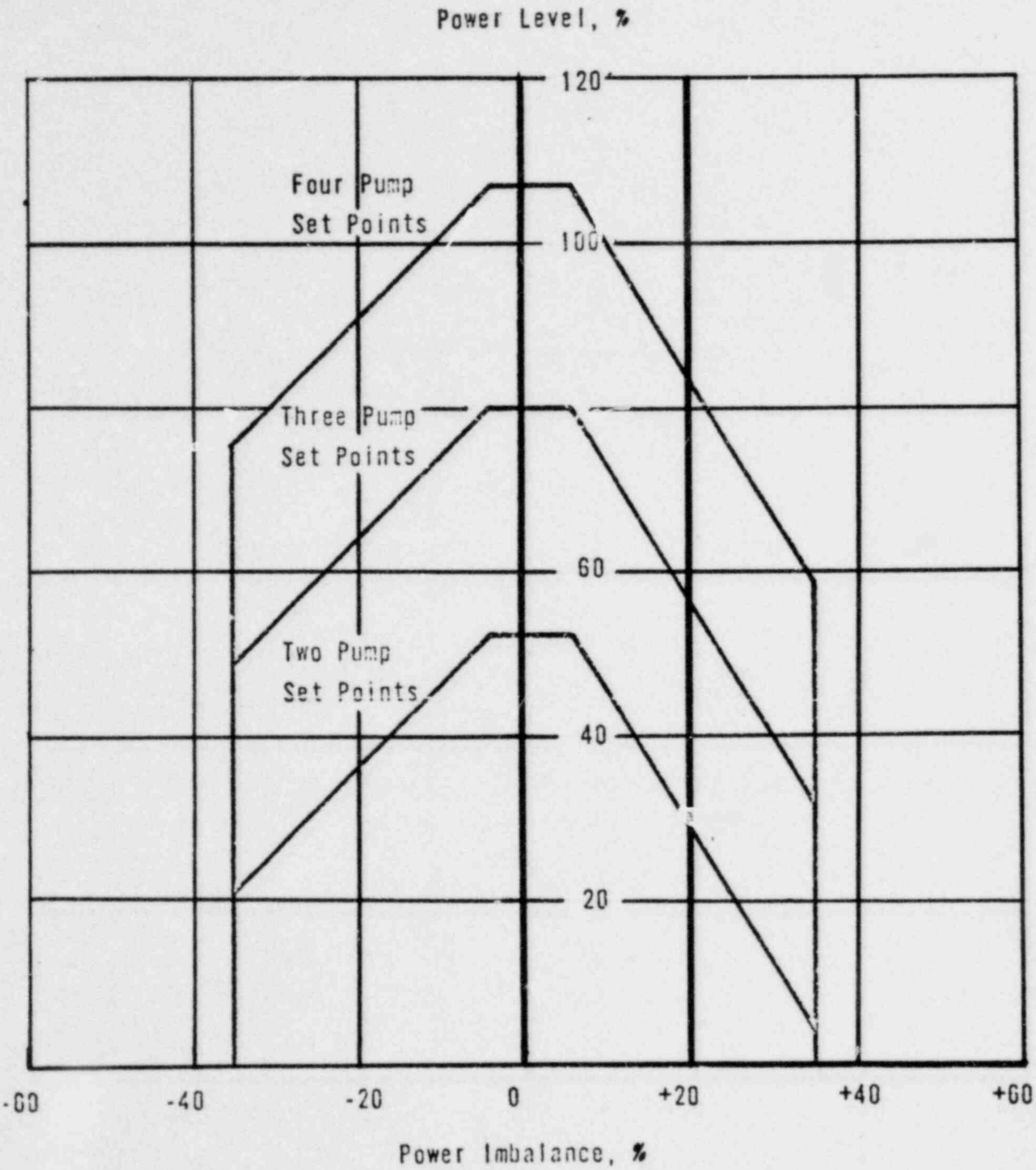
(3) FSAR, Section 14.1.2.7

(4) FSAR, Section 14.1.2.8

(5) FSAR, Section 14.1.2.6



ARKANSAS POWER & LIGHT CO. ARKANSAS NUCLEAR ONE-UNIT 1	PROTECTIVE SYSTEM MAXIMUM ALLOWABLE SET POINT	FIG. NO. 2.3-1



ARKANSAS POWER & LIGHT CO. ARKANSAS NUCLEAR ONE-UNIT 1	PROTECTIVE SYSTEM MAXIMUM ALLOWABLE SET POINTS	FIG.NO. 2-3-2

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	105.5	105.5	105.5	5.0(3)
2. Nuclear power based on flow(2) and imbalance, % of rated, max	1.07 times flow minus reduction due to imbalance(s)	1.07 times flow minus reduction due to imbalance(s)	1.07 times flow minus reduction due to imbalance(s)	Bypassed
3. Nuclear power based on pump monitors, % of rated, max(5)	NA	NA	55%	Bypassed
4. High reactor coolant system pressure, psig, max	2355	2355	2355	1720(4)
5. Low reactor coolant system pressure, psig, min	1800	1800	1800	Bypassed
6. Variable low reactor coolant system pressure, psig, min	$(16.25T_{out}-7756)$(1)	$(16.25T_{out}-7756)$(1)	$(16.25T_{out}-7756)$(1)	Bypassed
7. Reactor coolant temp, F, max	619	619	619	619
8. High reactor building pressure, psig, max	4 (18.7 psia)	4 (18.7 psia)	4 (18.7 psia)	4 (18.7 psia)

(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 during two-pump operation.

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 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.

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.

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 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 a rod withdrawal accident.[5] The pressurizer code safety valve lift set point shall be set at 2500 psig ± 1 percent allowance for error and each valve shall be capable of relieving 300,000 lb/h of saturated steam at a pressure not greater than 3 percent above the set pressure.

REFERENCES

(1) FSAR, Tables 9-10 and 4-3 through 4-7

(2) FSAR, Section 4.2.5.1 and 9.5.2.3

(3) FSAR, Section 4.2.5.4

(4) FSAR, Section 4.3.10.4 and 4.2.4

(5) FSAR, Section 4.3.7

3.1.2 Pressurization, Heatup, and Cooldown Limitations

Specification

3.1.2.1 Hydro Tests:

For thermal steady state system hydro tests the system may be pressurized to the limits set forth in Specification 2.2 when there are fuel assemblies in the core and to ASME Code Section III limits when no fuel assemblies are present provided:

a. Prior to initial criticality the reactor coolant system temperature is 100°F or greater or

b. After initial criticality and during the first two years of operation the reactor coolant system temperature is 215°F or greater.

3.1.2.2 Leak Tests

a. Leak tests may be conducted under the provisions of 3.1.2.1 above or

b. After initial criticality and during the first two years of operation the system may be tested to a pressure of 1150 psig provided that the system temperature is 175°F or greater.

3.1.2.3 For the first two years of power operation (1.7×10^6 thermal megawatt days) the reactor coolant pressure and the system heatup and cooldown rates(with the exception of the pressurizer shall be limited in accordance with Figure 3.1.2-1 and Figure 3.1.2-2, and are as follows:

Heatup:

Allowable combinations of pressure and temperature shall be to the right of and below the limit line in Figure 3.1.2-1. The heatup rates shall not exceed those shown on Figure 3.1.2-1.

Cooldown:

Allowable combinations of pressure and temperature for a specific cooldown shall be to the left of and below the limit line in Figure 3.1.2-2. Cooldown rates shall not exceed those shown in Figure 3.1.2-2.

3.1.2.4 The secondary side of the steam generator shall not be pressurized above 200 psig if the temperature of the steam generator shell is below 100°F.

3.1.2.5 The pressurizer heatup and cooldown rates shall not exceed 100 F/hr. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 430°F.

3.1.2.6 Within two years of power operation, Figures 3.1.2-1 and 3.1.2-2 shall be updated.

Bases

All reactor coolant system components are designed to withstand the effects of cyclic loads due to system temperature and pressure changes.[1] These cyclic loads are introduced by unit load transients, reactor trips, and unit heatup and cooldown operations. The number of thermal and loading cycles used for design purposes are shown in Table 4-8 of the FSAR. The maximum unit heatup and cooldown rate of 100 F per hour satisfies stress limits for cyclic operation.(2) The 200 psig pressure limit for the secondary side of the steam generator at a temperature less than 100 F satisfies stress levels for temperatures below the DTT.(3) The plate material and welds in the core region of the reactor vessel have been tested to verify conformity to specified requirements and a maximum NDTT value of 10 F has been determined based on Charpy V-notch tests. The maximum NDTT value obtained for the steam generator shell material and welds was 40 F.

Figures 3.1.2-1 and 3.1.2-2 contain the limiting reactor coolant system pressure-temperature relationship for operation at DTT[(4)] and below to assure that stress levels are low enough to preclude brittle fracture. These stress levels and their bases are defined in Section 4.3.3 of the FSAR.

As a result of fast neutron irradiation in the region of the core, there will be an increase in the NDTT with accumulated nuclear operation. The predicted maximum NDTT increase for the 40-year exposure is shown on Figure 4-10.[(4)] The actual shift in NDTT will be determined periodically during plant operation by testing of irradiated vessel material samples located in this reactor vessel.[(5)] The results of the irradiated sample testing will be evaluated and compared to the design curve (Figure 4-11 of FSAR) being used to predict the increase in transition temperature.

The design value for fast neutron (E > 1 Mev) exposure of the reactor vessel is 3.0×10^{10} n/cm^2-sec at 2568 MWt rated power and an integrated exposure of 3.0×10^{19} n/cm^2 for 40 years operation.[(6)] The calculated maximum values are 2.2×10^{10} n/cm^2-sec and 2.2×10^{19} n/cm^2 integrated exposure for 40 years operation at 80 percent load.[(4)] Figure 3.1.2-1 is based on the design value which is considerably higher than the calculated value. The DTT value for Figure 3.1.2-1 is based on the projected NDTT at the end of the first two years of operation. During these two years, the energy output has been conservatively estimated to be 1.7×10^6 thermal megawatt days which is equivalent to 655 days at 2568 MWt core power. The projected fast neutron exposure of the reactor vessel for the two years is 1.7×10^{18} n/cm^2 which is based on the 1.7×10^6 thermal megawatt days and the design value for fast neutron exposure.

The actual shift in NDTT will be established periodically during plant operation by testing vessel material samples which are irradiated cumulatively by securing them near the inside wall of the vessel in the core area. To compensate for the increases in the NDTT caused by irradiation, the limits on the pressure-temperature relationship are periodically changed to stay within the established stress limits during heatup and cooldown.

The NDTT shift and the magnitude of the thermal and pressure stresses are sensitive to integrated reactor power and not to instantaneous power level. Figures 3.1.2-1 and 3.1.2-2 are applicable to reactor core thermal ratings up to 2568 MWt.

The pressure limit line on Figure 3.1.2-1 has been selected such that the reactor vessel stress resulting from internal pressure will not exceed 15 percent yield strength considering the following:

A. A 25 psi error in measured pressure.

B. System pressure is measured in either loop.

C. Maximum differential pressure between the point of system pressure measurement and reactor vessel inlet for all operating pump combinations.

For adequate conservatism, in lieu of portions of the Fracture Toughness Testing Requirements of the proposed Appendix G to 10 CFR 50, a maximum pressure of 550 psig and a maximum heatup rate of 50 F/hr has been imposed below 275 F as shown on Figure 3.1.2-1.

The spray temperature difference restriction based on a stress analysis of the spray line nozzle is imposed to maintain the thermal stresses at the pressurizer spray line nozzle below the design limit. Temperature requirements for the steam generator correspond with the measured NDTT for the shell.

REFERENCES

(1) FSAR, Section 4.1.2.4

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

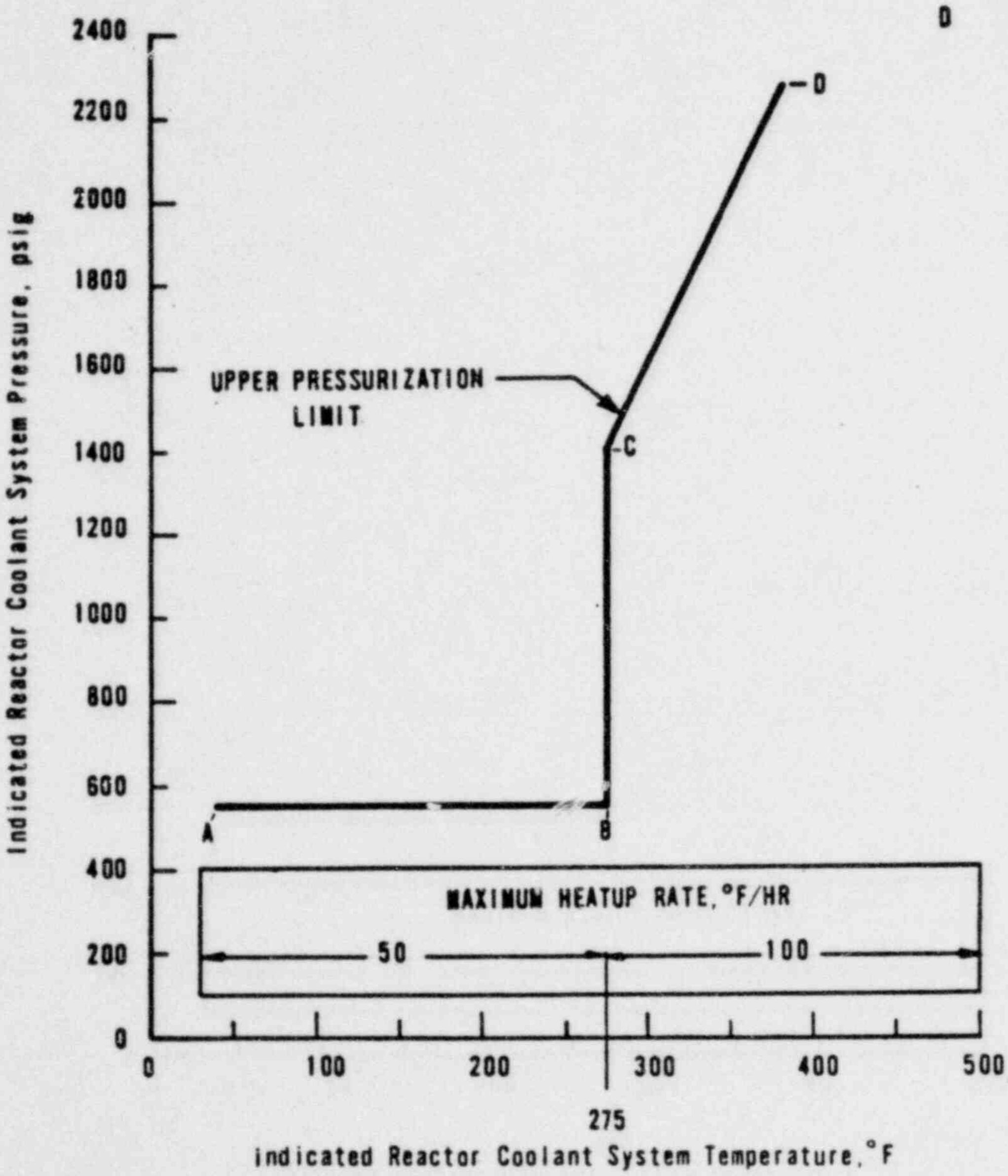
(3) FSAR, Section 4.3.10.5

(4) FSAR, Section 4.3.3

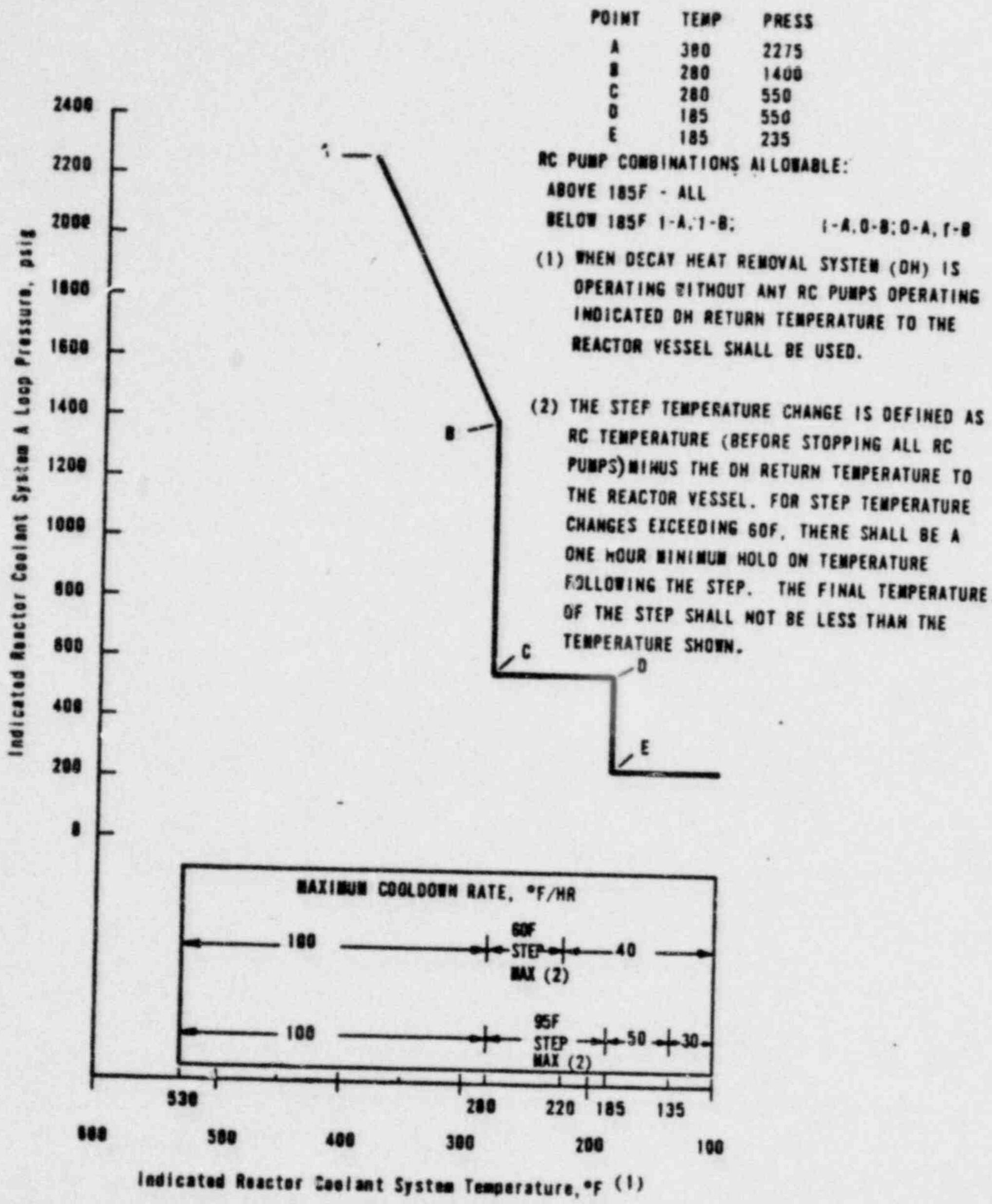
(5) FSAR, Section 4.4.5

(6) FSAR, Sections 4.1.2.8 and 4.3.3

POINT	TEMP	PRESS.
A	40	550
B	275	550
C	275	1400
D	380	2275



ARKANSAS POWER & LIGHT CO. ARKANSAS NUCLEAR ONE-UNIT 1	REACTOR COOLANT SYSTEM HEATUP LIMITATIONS (APPLICABLE UP TO AN INTEGRATED EXPOSURE OF 1.7×10^{18} n/cm^2 OR DTT = 134°F)	FIG. NO. 3.1.2-1



ARKANSAS POWER & LIGHT CO. ARKANSAS NUCLEAR ONE-UNIT 1	REACTOR COOLANT SYSTEM COOLDOWN LIMIT (APPLICABLE UP TO DTT = 175°F)	FIG. NO. 3.1.2-2

3.1.3 Minimum Conditions For Criticality

Specification

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 45 and 305 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.

Bases

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 2100 psia to saturation pressure of 900 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.

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.

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, Section 3.2.2.1.5

3.1.4 Reactor Coolant System Activity

Specification

3.1.4.1 The total activity of the reactor coolant due to nuclides with half lives longer than 30 minutes shall not exceed $40/\bar{E}$ microcuries per milliliter whenever the reactor is critical. E is the average (mean) bata and gamma energies per disintegration, in MeV weighted in proporation 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 2763 ft^3 of hot reactor coolant is assumed to have leaked into the secondary system.

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 be less than 10% of 10 CFR 100 should a steam generator tube rupture accident occur.

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 assumed that meteorological conditions during the course of the accident correspond to a X/Q value of 6.5×10^{-4} sec/m^3, which will not be exceeded more than 5% of the time.

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

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

$$A_{max}(\mu c/cc) = \frac{(\text{Dose})_{max}}{0.246 \cdot \bar{E} \cdot V \cdot X/Q} = \frac{0.5}{0.246 \times \bar{E} \times 78.25 \times 6.5 \times 10^{-4}}$$

$$A_{max}(\mu c/cc) = 40 / \bar{E}$$

where

A = reactor coolant specific activity at 580°F ($\mu Ci/ml = Ci/m^3$),

V = reactor coolant hot volume leaked into secondary system (2763 ft^3 = 78.25 m^3),

X/Q = atmospheric dispersion coefficient at exclusion distance for a two hour period (6.5×10^{-4} s/m^3),

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

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

A. Quantitative measurement of the specific activity (in units of $\mu Ci/cc$) 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 continuous 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.

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).

Bases - (Continued)

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 exchanger 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.

3.1.6.2 If unidentified reactor coolant leakage (exceeding normal evaporative losses) 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 it is determined that any reactor coolant leakage exists through a non-isolable fault in a reactor coolant system strength boundary (such as the reactor vessel, piping, valve body, etc. except steam generator tubes), the reactor shall be shutdown and a 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 3.1.6.1, 3.1.6.2, or 3.1.6.3, the rate of cooldown and the conditions of shutdown shall be determined by the safety evaluation for each case and reported as required by Specification 6.12.3.

3.1.6.5 Action to evaluate the safety implication of reactor coolant leakage shall be initiated within 4 hours of detection. 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 offsite personnel to radiation is within the guidelines of 10 CFR 20.

3.1.6.6 If reactor shutdown is required per Specification 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 When the reactor is at power operation, 3 reactor coolant leak detection systems of different operating principles shall be in operation. One of these systems is sensitive to radioactivity and consists of a radioactive gas detector and an air particulate activity detector. Both of these instruments may be out-of-service simultaneously for a period of no more than 72 hours provided 2 other means are available to detect leakage and reactor building air samples are taken and analyzed in the laboratory at least once per shift.

3.1.6.8 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 coolant system shall not be considered as reactor coolant leakage and shall not be subject to the consideration of Specifications 3.1.6.1 - 3.1.6.6 except that such losses when added to leakage shall not exceed 30 gpm.

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 prevent a large leak from masking the presence of a smaller leak. Reactor building sump level, 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 radioactive contamination and cleanup or it could develop into a still more serious problem; and therefore, first indications of such leakage will be followed up as soon as practicable.

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

When the source of leakage has been identified, the situation can be evaluated to determine if operation can safely continue. This evaluation will be performed by the 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 makeup pump and makeup would be available even during a loss of off-site power.

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

a. Leakage is monitored by a level indicator in the reactor building sump. Changes in normal sump level may be indicative of leakage from any of the systems located inside the reactor building such as the reactor coolant system, service water system, intermediate cooling system and steam and feedwater lines or condensation of humidity within the reactor building atmosphere. The reactor building sump contains 63.6 gallons per inch of height. A 1 gpm leak would be detected in less than 1 hour.

b. Total reactor coolant system leakage rate is periodically determined by comparing indications of reactor power, reactor coolant temperature, pressurizer water level and reactor coolant 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 reactor coolant makeup tank resulting in a tank level decrease. The reactor coolant makeup tank capacity is 31 gallons per inch of height and each graduation on the level recorder represents 2 inches of tank height. This inventory monitoring method is capable of detecting changes on the order of 62 gallons. A 1 gpm leak would therefore be detectable within approximately 1.1 hours.

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

c. The reactor building gaseous monitor is sensitive to low leak rates if expected values of failed fuel exist. The rates of reactor coolant leakage to which the instrument is sensitive are discussed in FSAR Section 4.2.3.8.

The upper limit of 30 gpm is based on the contingency of a hypothetical loss of all AC power. A 30 gpm loss of water in conjunction with a hypothetical loss of all AC power and subsequent cooldown of the reactor coolant system by the atmospheric dump system and steam driven emergency 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 both electrical power to the station and makeup flow to the reactor coolant system.

References

FSAR Section 4.2.3.8

3.1.7. Moderator Temperature Coefficient of Reactivity

Specification

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

Bases

A non-positive moderator coefficient at power levels above 95% of rated power is specified such that the maximum clad temperatures will not exceed the Interim Acceptance Criteria based on LOCA analyses. Below 95% of rated power the Interim Acceptance Criteria will not be exceeded with a positive moderator temperature coefficient of $+0.5 \times 10^{-4}\ \Delta k/k/^{\circ}F$ corrected to 95% of rated power. All other accident analyses as reported in the FSAR have been performed for a range of moderator temperature coefficients including $+0.5 \times 10^{-4}\ \Delta k/k/^{\circ}F$.

When the hot zero-power value is corrected to obtain the hot 95% value, the following corrections will be applied.

1. Uncertainty in isothermal measurement — The measured moderator temperature coefficient will contain uncertainty owing to (a) $\pm 0.2^{\circ}F$ in the ΔT of the base and perturbed conditions, and (b) uncertainty in the reactivity measurement of $\pm 0.1 \times 10^{-4}\ \Delta k/k$.

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

2. Doppler coefficient 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 for the moderator. The measured temperature coefficient must be increased by $0.16 \times 10^{-4}\ (\Delta k/k)/^{\circ}F$ to obtain a pure moderator temperature coefficient.

3. Moderator temperature change — The hot zero-power measurement must be reduced by $0.08 \times 10^{-4}\ \Delta k/k/^{\circ}F$. This corrects for the difference in water temperature from zero power (532F) and 15% power (580F). Above this power, the average moderator temperature remains 580F.

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). This correction is $-0.0022 \times 10^{-4}\ \Delta\alpha_m/\Delta\%$ power. It adjusts the moderator coefficient at 15% power to the coefficient at any power level above 15%. For example, the power effect adjustment from a 15% coefficient to 100% power is

$$(-0.0022 \times 10^{-4})(100\% - 15\%) = -0.187 \times 10^{-4}\ \Delta\alpha_m.$$

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. The correction is $0.15 \times 10^{-6}\ \Delta\alpha_m/\Delta ppm$. (The magnitude of the correction varies slightly with boron concentration; the value presented above is valid for the range of from 1000 to 1400 ppm.

6. Control rod insertion — This correction is for the difference in moderator coefficients between an unrodded and rodded core. The moderator coefficient should be reduced by $0.09 \times 10^{-4}\ \Delta k/k/^{\circ}F$ to adjust for the nominal rod insertion pattern at full power.

7. Isothermal to distributed temperatures — The correction for spatially distributed moderator temperature effects has been found to be less than or equal to zero. Therefore, zero is a conservative correction value for distributed effects.

REFERENCES

[1]FSAR, Section 14.

[2]FSAR, Section 3.

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 1720 psig shutdown bypass trip setting limits shall apply in accordance with Table 2.3-1.

B. Above 1800 psig nuclear overpower trip shall be set at less than 5.0 percent. Other settings shall be in accordance with Table 2.3-1.

3.1.8.2 Startup rate rod withdrawal hold[1] 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%Δk/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.

REFERENCES

(1) FSAR, Section 7.2.2.1.3.

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/liter 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/liter 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.

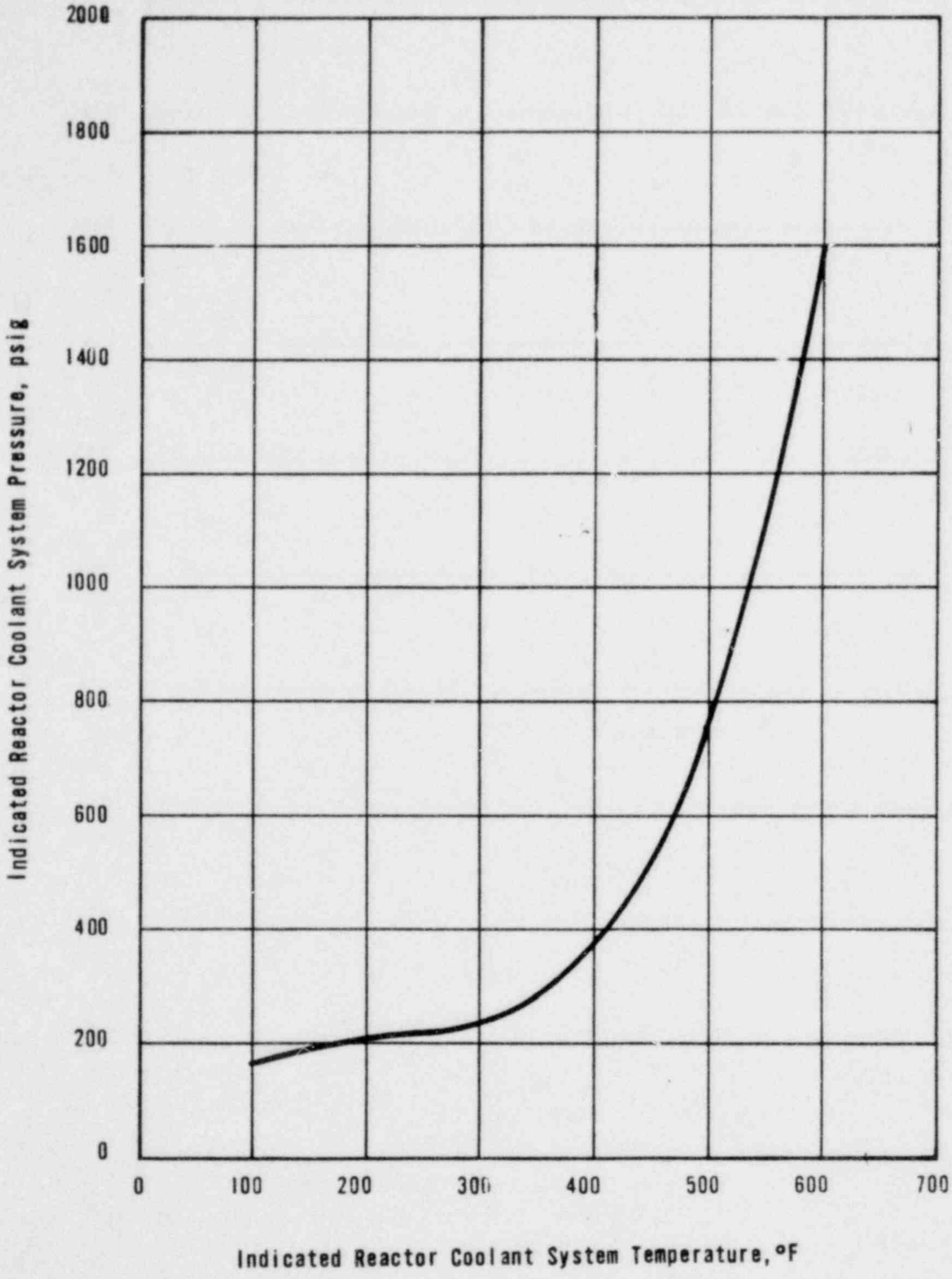
Bases

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/liter of water, the dissolved gas concentration is conservatively assumed to be 100 std. cc/liter 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/liter 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/liter 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.



ARKANSAS POWER & LIGHT CO. ARKANSAS NUCLEAR ONE-UNIT 1	LIMITING PRESSURE VS TEMPERATURE FOR CONTROL ROD DRIVE OPERATION WITH 100 STD CC/LITER H_2O	FIG. NO. 3.1.9-1

3.2 MAKEUP AND CHEMICAL ADDITION SYSTEMS

Applicability

Applies to the operational status of the makeup 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

3.2.1 The reactor shall not be heated or maintained above 200°F unless the following conditions are met:

3.2.1.1 Two makeup pumps are operable except as specified in Specification 3.3.

3.2.1.2 A source of concentrated boric acid solution in addition to that in the borated water storage tank is available and operable. This requirement is fulfilled by the boric acid addition tank. This tank shall contain at least the equivalent of 47 inches ($550 ft^3$) of 8700 ppm boron as boric acid solution with a temperature of at least 10°F above the crystallization temperature. System piping and valves necessary to establish a flow path from the tank to the makeup system shall also be operable and shall have at least the same temperature as the boric acid addition tank. One associated boric acid pump is operable.

3.2.1.3 The boric acid addition tank and associated piping, valves and both pumps may be out of service for a maximum of 24 hours. After the 24 hour period, if the system is not returned to service and operable, the reactor shall be brought to the hot shutdown condition within an additional 12 hours.

Bases

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

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% subcritical margin in the cold condition at the worst time in core life with a stuck control rod assembly.

Minimum volumes (including a 10% safety factor) of 550 ft^3 of 8700 ppm boron as boric acid solution in the boric acid addition tank or 16,000 gallons of 2270 ppm boron as boric acid solution in the borated water storage tank(3) will each satisfy this 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 principal method of adding boron to the primary system is to pump the concentrated boric acid solution (8700 ppm boron, minimum) into the makeup tank using the 25 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 three hours. The alternate method of addition is to inject boric acid from the borated water storage tank using the makeup pumps. the required 16,000 gallons of boric acid can be injected in less than two hours using only one of the makeup pumps.

Concentration of boron in the boric acid addition 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 10°F above the crystallization temperature for the concentration present. Once in the makeup system, the concentrate is sufficiently well mixed and diluted so that normal system temperatures assure boric acid solubility.

REFERENCES

(1) FSAR, Section 9.1; 9.2

(2) FSAR, Figure 6-2

(3) FSAR, Section 3.3

3.3 EMERGENCY CORE COOLING, REACTOR BUILDING COOLING, REACTOR BUILDING SPRAY AND PENETRATION ROOM VENTILATION SYSTEMS

Applicability

Applies to the emergency core cooling, reactor building cooling, reactor buildi.g spray and reactor building penetration room ventilation systems.

Objectivity

To define the conditions necessary to assure immediate availability of the emergency core cooling, reactor building cooling, reactor building spray and reactor building penetration room ventilation systems.

Specification

3.3.1 The following equipment shall be operable whenever containment integrity is established as required by Specification 3.6.1:

(A) One reactor building spray pump and its associated spray nozzle header.

(B) Two reactor building cooling fans and associated cooling units.

(C) Two out of three service water pumps shall be operable, powered from independent essential buses, to provide redundant and independent flow paths.

(D) Reactor building penetration room ventilation systems consisting of both penetration room fans and their associated filters. All manually operated system valves shall be locked open.

(E) Two engineered safety feature acutated low pressure injection pumps shall be operable.

(F) Both low pressure injection coolers and their cooling water supplies shall be operable.

(G) Two BWST level instrument channels shall be operable.

(H) The borated water storage tank shall contain a minimum level of 35.9 feet (350,000 gallons) of water having a minimum concentration of 2270 ppm boron at a temperature not less than 40°F. The manual valve on the dischage line from the borated water storage tank shall be locked open.

(I) The four reactor building emergency sump isolation valves to the LPI system shall be either manually or remote-manually operable.

(J) The engineered safety features valves associated with each of the above systems shall be operable or locked in the ES position.

3.3.2 In addition to 3.3.1 above, the following ECCS equipment shall be operable when the reactor coolant system is above 350°F and irradiated fuel is in the core:

(A) Two out of three high pressure injection (makeup) pumps shall be maintained operable, powered from independent essential busses, to provide redundant and independent flow paths.

(B) Engineered safety features valves associated with 3.3.2.a above shall be operable or locked in the ES position.

3.3.3 In addition to 3.3.1 and 3.3.2 above, the following ECCS equipment shall be operable when the reactor coolant system is above 800 psig.

(A) The two core flooding tanks shall each contain an indicated minimum of 13 $\pm$ 0.4 feet (1040 $\pm$ 30 ft^3) of borated water at 600 $\pm$ 25 psig.

(B) Core flooding tank boron concentration shall not be less than 2270 ppm boron.

(C) The electrically operated discharge valves from the core flood tanks shall be open and breakers locked open and tagged.

(D) One of the two pressure instrument channels and one of the two level instrument channels per core flood tank shall be operable.

3.3.4 The reactor shall not be made cricical unless the following equipment in addition to 3.3.1, 3.3.2, and 3.3.3 above is operable.

(A) Two reactor building spray pumps and their associated spray nozzle headers and four reactor building emergency cooling fans and associated cooling units.

(B) The sodium thiosulfate tank shall contain an indicated 31 ft of 30 wt% solution sodium thiosulfate (37,500 lb). The sodium hydroxide tank shall contain an indicated 31 ft. of 20 wt% solution sodium hydroxide (20,500 lb.).

(C) All manual valves in the main discharge lines of the sodium thiosulfate and sodium hydroxide tanks shall be locked open.

(D) Engineered safety feature valves and interlocks associated with 3.3.1, 3.3.2, and 3.3.3 shall be operable or locked in the ES position.

3.3.5 Except as noted in 3.3.6 below, maintenance shall be allowed during power operation on any component(s) in the high pressure injection, low pressure injection, service water, reactor building spray, reactor building cooling, penetration room ventilation, CFT pressure and level instrumentation, and BWST level instrumentation systems which will not remove more than one train of each system from service. Components shall not be removed from service so that the affected system train is inoperable for more than 24 consecutive hours. If the system is not restored to meet the requirements of Specification 3.3.1, 3.3.2, 3.3.3, or 3.3.4 within 24 hours the reactor shall be placed in the hot shutdown condition within 12 hours. If the requirements of Specification 3.3.1, 3.3.2, 3.3.3, or 3.3.4 are not met within an additional 48 hours, the reactor shall be placed in the cold shutdown condition within 24 hours.

3.3.6 Exceptions to 3.3.5 shall be as follows:

(A) Both core flooding tanks shall be operational above 800 psig.

(B) Both motor-operated valves associated with the core flooding tanks shall be fully open above 800 psig.

(C) One of the two pressure instrument channels and one of the two level instrument channels per core flood tank shall be operable.

3.3.7 Prior to initiating maintenance on any of the components, the duplicate (redundant) component shall be tested or have been tested within 24 hours to assure operability.

Bases

The requirements of Specification 3.3.1 assure that below 350°F, adequate long term core cooling is provided. Two low pressure injection pumps are specified. However, only one is necessary to supply emergency coolant to the reactor in the event of a loss-of-coolant accident.

The post-accident reactor building cooling and long-term pressure reduction may be accomplished by four cooling units, by two spray units or by a combination of two cooling units and one spray unit. Post-accident iodine removal may be accomplished by one of the two spray system strings. The specified requirements assure that the required post-accident components are available for both reactor building cooling and iodine removal. Specification 3.3.1 assures that the required equipment is operational.

The borated water storage tank is used for three purposes:

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

(B) As an alternate supply of borated water for reaching cold shutdown.(2)

(C) As a supply of borated water for flooding the fuel transfer canal during refueling operation.(3)

350,000 gallons of borated water are supplied for emergency core cooling and reactor building spray in the event of a loss-of-coolant accident. This amount fulfills requirements for emergency core cooling. 16,000 gallons of borated water are required to reach cold shutdown. The borated water storage tank capacity of 380,000 gallons is based on refueling volume requirements. Heaters maintain the borated water supply at a temperature to prevent crystallization and local freezing of the boric acid. The boron concentration is set at a value that will maintain the core at least 1 percent $\Delta k/k$ subcritical at 70°F without any control rods in the core. The concentration for 1% $\Delta k/k$ subcriticality is 1609 ppm boron in the core, while the minimum value specified in the borated water storage tank is 2270 ppm boron.

Specification 3.3.2 assures that above 350°F two high pressure injection pumps are also available to provide injection water as the energy of the reactor coolant system is increased.

Specification 3.3.3 assures that above 800 psig both core flooding tanks are operational. Since their design pressure is 600 ± 25 psig, they are not brought into the operational state until 800 psig to prevent spurious injection of borated water. Both core flooding tanks are specified as a single core flood tank has insufficient inventory to reflood the core.[1]

Specification 3.3.4 assures that prior to going critical the redundant reactor building cooling unit and spray are operational.

The spray system utilizes common suction lines with the low pressure injection 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 3.3.5 and 3.3.6 provided requirements in Specification 3.3.7 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 24 hours is acceptable if the operability of equipment redundant to that removed from service is demonstrated within 24 hours prior to removal.

In the event that the need for emergency core cooling should occur, functioning of one train (one high pressure injection pump, one low pressure injection pump, and both core flooding tanks) will protect the core and in the event of a main coolant loop severence, limit the peak clad temperature to less than 2300°F and the metal-water reaction to that representing less than 1 percent of the clad.

The service water system consists of two independent but interconnected, full capacity, 100% redundant systems, to ensure continuous heat removal.[4]

One service water pump is required for normal operation. The normal operating requirements are greater than the emergency requirements following a loss-of-coolant accident.

The penetration room ventilation system consists of two independent, full capacity, 100% redundant trains. If one train is removed from operation, the other train must be operable.[5]

REFERENCES

(1) FSAR, Section 14.2.5

(2) FSAR, Section 3.2

(3) FSAR, Section 9.5.2

(4) FSAR, Section 9.3.1

(5) FSAR, Section 6.5

3.4 STEAM AND POWER CONVERSION SYSTEM

Applicability

Applies to the turbine cycle components for removal of reactor decay heat.

Objective

To specify minimum conditions of the turbine cycle equipment necessary to assure the capability to remove decay heat from the reactor core.

Specifications

3.4.1 The reactor shall not be heated, above 280°F unless the following conditions are met:

1. Capability to remove a decay heat load of 5% full reactor power by at least one of the following means:

    a. A condensate pump and a main feedwater pump, using turbine by-pass valve.

    b. A condensate pump and the auxiliary feedwater pump using turbine by-pass valve.

2. Fourteen of the steam system safety valves are operable.

3. A minimum of 16.3 ft. (107,000 gallons) of water is available in the condensate storage tank.

4. Both emergency feedwater pumps are operable.

5. Both main steam block valves and both main feedwater isolation valves are operable.

6. The emergency feedwater valves associated with specification 3.4.1.4 shall be operable.

3.4.2 Components required by Specification 3.4.1 to be operable shall not be removed from service for more than 24 consecutive hours. If the system is not restored to meet the requirements of Specification 3.4.1 within 24 hours the reactor shall be placed in the hot shutdown condition within 12 hours. If the requirements of Specification 3.4.1 are not met within an additional 48 hours, the reactor shall be placed in the cold shutdown condition within 24 hours.

Bases

The feedwater flow required to remove decay heat corresponding to 5% full power with saturated steam at 1065 psia (lowest setting of steam safety valve) as a function of feedwater temperature is:

Feedwater Temperature	Flow
60	758
90	777
120	799
140	814

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 either a main or the auxiliary feedwater pump.

In the incredible event of loss of all AC power, feedwater is supplied by the turbine driven emergency feedwater pump which takes suction from the condensate storage tank. Decay heat is removed from a steam generator by steam relief through the atmospheric dump valves or safety valves. Fourteen of the steam system safety valves will relieve the necessary amount of steam for rated reactor power.

The minimum amount of water in the condensate storage tank would be adequate for about 4.5 hours of operation. This is based on the estimate of the average emergency flow to a steam generator being 390 gpm. This operation time with the volume of water specified would not be reached, since the decay heat removal system would be brought into operation within 4 hours or less.

If the turbine driven emergency feedwater pump has not been verified to be operable within 3 months prior to heatup its operability will be verified upon reaching hot shutdown conditions.

References

FSAR, Section 10

3.5 INSTRUMENTATION SYSTEMS

3.5.1 Operational Safety Instrumentation

Applicability

Applies to unit instrumentation and control systems.

Objectives

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 3 and 4 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 3 and 4, operation shall be limited as specified in Column 5.

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. Only one channel bypass key shall be accessible for use in the control room.

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 instruments come 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.

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 following detection. The condition will be corrected and the remaining trip devices shall be tested within eight hours following detection. 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 the requirements of Table 3.5.1-1, Columns 3 and 4, are met.

Operation at rated power is permitted as long as the systems have at least the redundancy requirements of Column 4 (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 to allow on-line testing or maintenance on only one channel at a time during power operation. Each channel is provided with alarm and lights to indicate when that channel is bypassed. There will be one reactor protection system channel bypass switch key permitted in the control room.

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.

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.

The ESAS employs three independent and identical analog channels, which supply trip signals to two independent, identical digital subsystems. In order to actuate the safeguards systems, two out of three analog channels must trip. This will cause both digital subsystems to trip. Tripping of either digital subsystem will actuate all safeguards systems associated with that digital subsystem.

Because only one digital subsystem is necessary to actuate the safeguards systems and these systems are capable of tripping even when they are being tested, a single failure in a digital subsystem cannot prevent protective action.

Removal of a module required for protection from a RPS channel will cause that channel to trip, unless that channel has been bypassed, so that only one channel of the other three must trip to cause a reactor trip. Thus, sufficient redundancy has been built into the system to cover this situation.

Removal of a module required for protective action, from an analog ESAS channel will cause that channel to trip, so that only one of the other two must trip to actuate the safeguards systems. Removal of a module required for protective action from a digital ESAS subsystem will not cause that subsystem to trip. The fact that a module has been removed will be continuously annunciated to the operator. The redundant digital subsystem is still sufficient to indicate complete ESAS action.

The testing schemes of both the RPS and the ESAS enable complete system testing while the reactor is operating. Each channel is capable of being tested independently so that operation of individual channels may be evaluated.

The Automatic Closure and Isolation System (ACI) is designed to close the Decay Heat Removal System (DHRS) return line isolation valves when the Reactor Coolant System (RCS) pressure exceeds a selected fraction of the DHRS design pressure or when core flooding system isolation valves are opened. The ACI is designed to permit manual operation of the DHRS return line isolation valves when permissive conditions exist. In addition, the ACI is designed to disallow manual operation of the valves when permissive conditions do not exist.

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 these 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. Four hours is ample time to test the remaining trip devices and in many cases make on-line repairs.

The Steam Line Break Instrumentation and Control System (SLBIC) is designed to automatically close the Main Steam Block valves and the Main Feedwater Isolation valves upon loss of pressure in any of the two main steam lines.

The SLBIC is also designed to be reset from its trip position only when the system is shut down or the Main Steam line pressure is below 600 psig.

REFERENCE

FSAR, Section 7.1

Table 3.5.1-1 Instrumentation Limiting Conditions for Operation
(Note 6)

REACTOR PROTECTION SYSTEM

Functional Unit	1 No. of channels	2 No. of channels for system trip	3 Min. operable channels	4 Min. degree of redundancy	5 Operator action is conditions of column 3 or 4 cannot be met
1. Manual pushbutton	1	1	1	0	Note 1
2. Power range instrument channel	4	2	3 (Note 4)	1 (Note 4)	Note 1
3. Intermediate range instrument channels	2	Note 7	1	0	Notes 1, 2
4. Source range instrument channels	2	Note 7	1	0	Notes 1, 2, 3
5. Reactor coolant temperature instrument channels	4	2	2	1	Note 1
6. Pressure-temperature instrument channels	4	2	2	1	Note 1
7. Flux/imbalance/flow instrument channels	4	2	2	1	Note 1
8. Reactor coolant pressure					
a. High reactor coolant pressure instrument channels	4	2	2	1	Note 1
b. Low reactor coolant pressure instrument channels	4	2	2	1	Note 1
9. Power/number of pumps instrument channels	4	2	2	1	Note 1
10. High reactor building pressure channels	4	2	2	1	Note 1

Table 3.5.1-1 (Contd)

ENGINEERED SAFEGUARDS ACTUATION SYSTEM	1	2	3	4	5
Functional Unit	No. of channels	No. of channels for system trip	Min. operable channels	Min. degree of redundancy	Operator action if conditions of column 3 or 4 cannot be met
1. High pressure injection system (Note 8)					
a. Reactor coolant pressure instrument channels	3	2	3 (Note 6)	1	Notes 1, 5
b. Reactor building 4 psig instrument channels	3	2	3 (Note 6)	1	Notes 1, 5
c. Manual trip pushbutton	2	1	2	1	Notes 1, 5
2. Low pressure injection system (Note 8)					
a. Reactor coolant pressure instrument channels	3	2	3 (Note 6)	1	Notes 1, 5
b. Reactor building 4 psig instrument channels	3	2	3 (Note 6)	1	Notes 1, 5
c. Manual trip pushbutton	2	1	2	1	Notes 1, 5
3. Reactor building isolation and reactor building cooling system (Note 8)					
a. Reactor building 4 psig instrument channel	3	2	3 (Note 6)	1	Notes 1, 5
b. Manual trip pushbutton	2	1	2	1	Notes 1, 5

Table 3.5.1-1 (Contd)

ENGINEERED SAFEGUARDS ACTUATION SYSTEM (Contd)	1	2	3	4	5
	No. of channels	No. of channels for system trip	Min. operable channels	Min. degree of redundancy	Operator action if conditions of column 3 or 4 cannot be met
4. Reactor building spray pumps (Note 8)					
a. Reactor building 30 psig instrument channel	3	2	3 (Note 6)	1	Notes 1, 5
b. Manual trip pushbutton	2	1	2	1	Notes 1, 5
5. Reactor building spray valves (Note 8)					
a. Reactor building 30 psig instrument channel	3	2	3 (Note 6)	1	Notes 1, 5
b. Manual trip pushbutton	2	1	2	1	Notes 1, 5
OTHER SAFETY RELATED SYSTEMS					
1. Decay heat removal system isolation valve automatic closure and interlock system					
a. Reactor coolant pressure instrument channels	2	1	2	1	Notes 1, 5
b. Core flood isolation valve interlocks	2	1	2	1	Notes 1, 5

Table 3.5.1-1 (Contd)

OTHER SAFETY RELATED SYSTEMS	1	2	3	4	5
Functional unit	No. of channels	No. of channels for system trip	Min. operable channels	Min. degree of redundancy	Operator action if conditions of column 3 or 4 cannot be met
2. Steam line break instrumentation control system (SLBIC)					
a. Main steam line instrument channels	2	1	2	1	Notes 1, 5

Notes:

1. Initiate a shutdown using normal operating instructions and place the reactor in the hot shutdown condition if the requirements of Columns 3 and 4 are not met within 12 hours.

2. When 2 of 4 power range instrument channels are greater than 10% rated power, hot shutdown is not required.

3. When 1 of 2 intermediate range instrument channels is greater than 10^{-10} amps, hot shutdown is not required.

4. 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, after which Note 1 applies.

5. If the requirements of Columns 3 or 4 cannot be met within an additional 48 hours, place the reactor in the cold shutdown condition within 24 hours.

6. The minimum number of operable channels may be reduced to 2, provided that the system is reduced to 1 out of 2 coincidence by tripping the remaining channel. Otherwise, specification 3.3 shall apply.

7. These channels initiate control rod withdrawal inhibits not reactor trips at <10% rated power. Above 10% rated power these inhibits are bypassed.

8. If any one component of a digital subsystem is inoperable, the entire digital subsystem is considered inoperable. Hence, the associated safety features are inoperable and Specification 3.3 applies.

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% Δk/k with the highest worth control rod fully withdrawn.

3.5.2.2 Operation with inoperable rods:

1. 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 groups shall not be permitted.

2. If a control rod in the regulating or safety rod groups is declared inoperable in the withdrawn position as defined in Specification 4.7.1.1 and 4.7.1.3, an evaluation shall be initiated immediately to verify the existance of 1% Δk/k available shutdown margin. Boration may be initiated either to the worth of the inoperable rod or until the regulating and transient rod groups are withdrawn to the limits of Specification 3.5.2.5.3, whichever occurs first. Simultaneously a program of exercising the remaining regulating and safety rods shall be initiated to verify operability.

3. If within one (1) hour of determination of an inoperable rod as defined in Specification 4.7.1, it is not determined that a 1% Δk/k available 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.

4. Following the determination of an inoperable rod as defined in Specification 4.7.1, all remaining rods shall be exercised within 24 hours and exercised weekly until the rod problem is solved.

5. 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% of the thermal power allowable for the reactor coolant pump combination.

6. 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 the thermal power allowable for the reactor coolant pump combination 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 3.5.2.5.3.

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

3.5.2.4 Quadrant tilt:

1. Except for physics tests, if quadrant tilt exceeds 4%, power shall be reduced immediately to below the power level cutoff (see Figures 3.5.2-1A and 3.5.2-1B). Moreover, the power level cutoff value shall be reduced 2% for each 1% tilt in excess of 4% tilt. For less than 4 pump operation, thermal power shall be reduced 2% of the thermal power allowable for the reactor coolant pump combination for each 1% tilt in excess of 4%.

2. Within a period of 4 hours, the quadrant power tilt shall be reduced to less than 4%, except for physics tests, or the following adjustments in setpoints and limits shall be made:

   a. The protection system maximum allowable setpoints (Figure 2.3-2) shall be reduced 2% in power for each 1% tilt.

   b. The control rod group withdrawal limits (Figures 3.5.2-1A and 3.5.2-1B) shall be reduced 2% in power for each 1% tilt in excess of 4%.

   c. The operational imbalance limits (Figure 3.5.2-3) shall be reduced 2% in power for each 1% tilt in excess of 4%.

3. If quadrant tilt is in excess of 25%, except for physics tests or diagnostic testing, the reactor will be placed in the hot shutdown condition. Diagnostic testing during power operation with a quadrant power tilt is permitted provided the thermal power allowable for the reactor coolant pump combination is restricted as stated in 3.5.2.4.1 above.

4. Quadrant tilt shall be monitored on a minimum frequency of once every two hours during power operation above 15% of rated power.

3.5.2.5 Control rod positions:

1. Technical Specification 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.

2. Operating rod group overlap shall be 25% $\pm$5 between two sequential groups, except for physics tests.

3. Except for physics tests or exercising control rods, the control rod withdrawal limits are specified on Figures 3.5.2-1A and 3.5.2-1B for four pump operation and on Figure 3.5.2-2 for three or two pump operation. If the 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 four hours.

4. Except for physics tests, power shall not be increased above the power level cutoff (see Figures 3.5.2-1) unless the xenon reactivity is within 10 percent of the equilibrium value for operation at rated power and asymptotically approaching stability.

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 tests, imbalance shall be maintained within the envelope defined by Figure 3.5.2-3. If the imbalance is not within the envelope defined by Figure 3.5.2-3, corrective measures shall be taken to achieve an acceptable imbalance. If an acceptable imbalance is not achieved within four 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.

Bases

The power-imbalance envelope defined in Figure 3.5.2-3 is based on LOCA analyses which have defined the maximum linear heat rate (see Figure 3.5.2-4) such that the maximum clad temperature will not exceed the Interim Acceptance Criteria. Corrective measures will be taken immediately should the indicated quadrant tilt, rod position, or imbalance be outside their specified boundary. Operation in a situation that would cause the interim acceptance criteria to be approached should a LOCA occur is highly improbable because all of the power distribution parameters (quadrant tilt, rod position, and imbalance) must be at their limits while simultaneously all other engineering and uncertainty factors are also at their limits.* Conservatism is introduced by application of:

a. Nuclear uncertainty factors
b. Thermal calibration
c. Fuel densification effects
d. Hot rod manufacturing tolerance factors

The 30 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 or banks defined as follows:

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

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

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

The minimum available rod worth provides for achieving hot shutdown by reactor trip at any time assuming, the highest worth control rod remains in the full out position.(1)

During power operation, inserted rod groups will not contain single rod worths greater than 0.65% $\Delta k/k$. This value has been shown to be safe by the safety analysis of the hypothetical rod ejection accident.(2) A single inserted control rod worth of 1.0% $\Delta k/k$ at beginning of life, hot, zero power would result in the same transient peak thermal power and therefore the same environmental consequences as a 0.65% $\Delta k/k$ ejected rod worth at rated power.

The quadrant power tilt limits set forth in Specification 3.5.2.4 have been established within the thermal analysis design base using the definition of quadrant power tilt given in Technical Specifications, Section 1.6. These limits in conjunction with the control rod position limits in Specification 3.5.2.5.3 ensure that design peak heat rate criteria are not exceeded during normal operation when including the effects of potential fuel densification.

The quadrant tilt and axial imbalance monitoring in Specifications 3.5.2.4.6 and 3.5.2.5.4, respectively, will normally be performed in the plant computer. The two-hour frequency for monitoring these quantities will provide adequate surveillance when the computer is out of service.

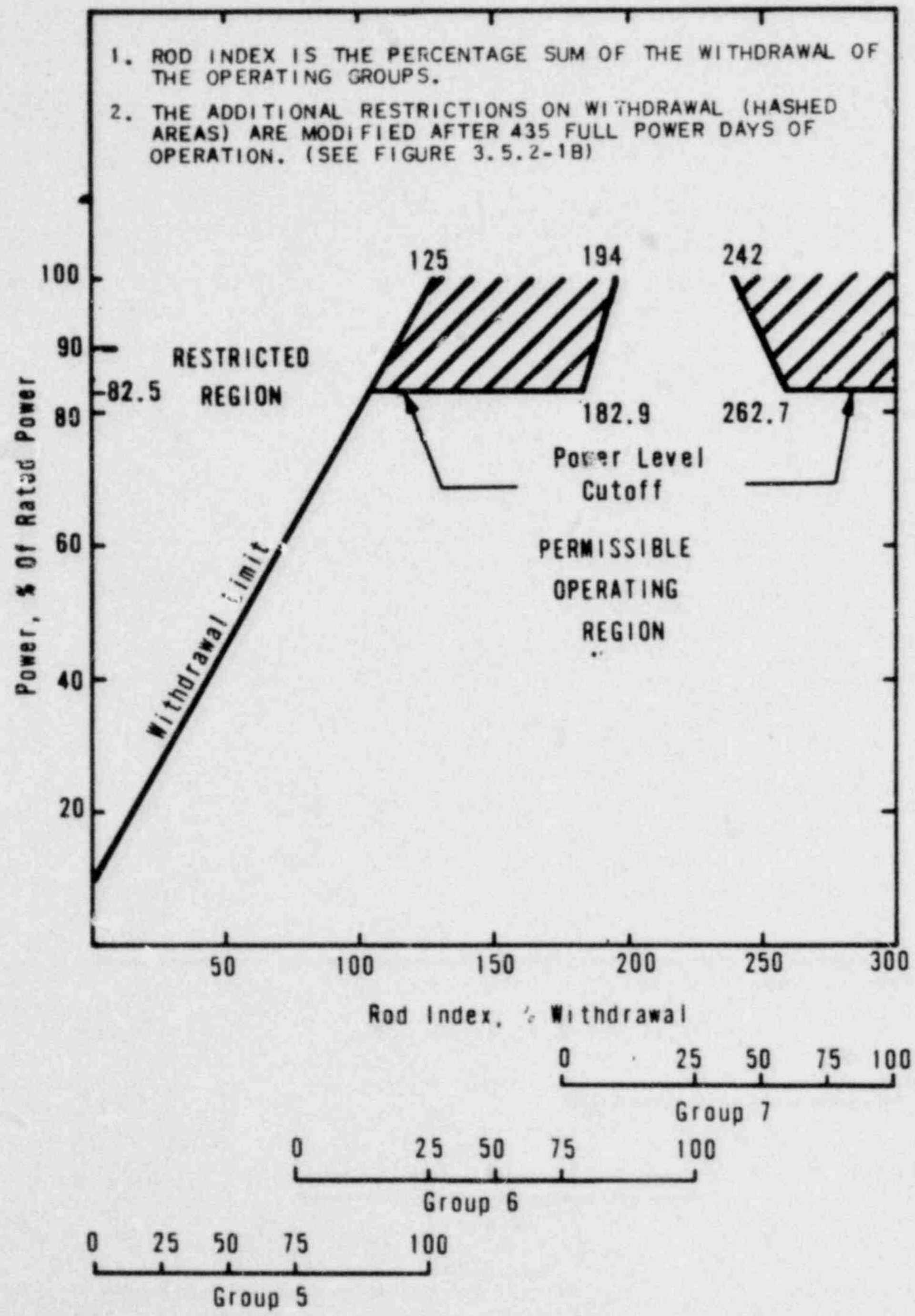
During the physics testing program, the high flux trip setpoints are administratively set as follows to ensure that an additional safety margin is provided:

Test Power	Trip Setpoint, %
0	<5
15	50
40	50
50	60
75	85
>75	105.5

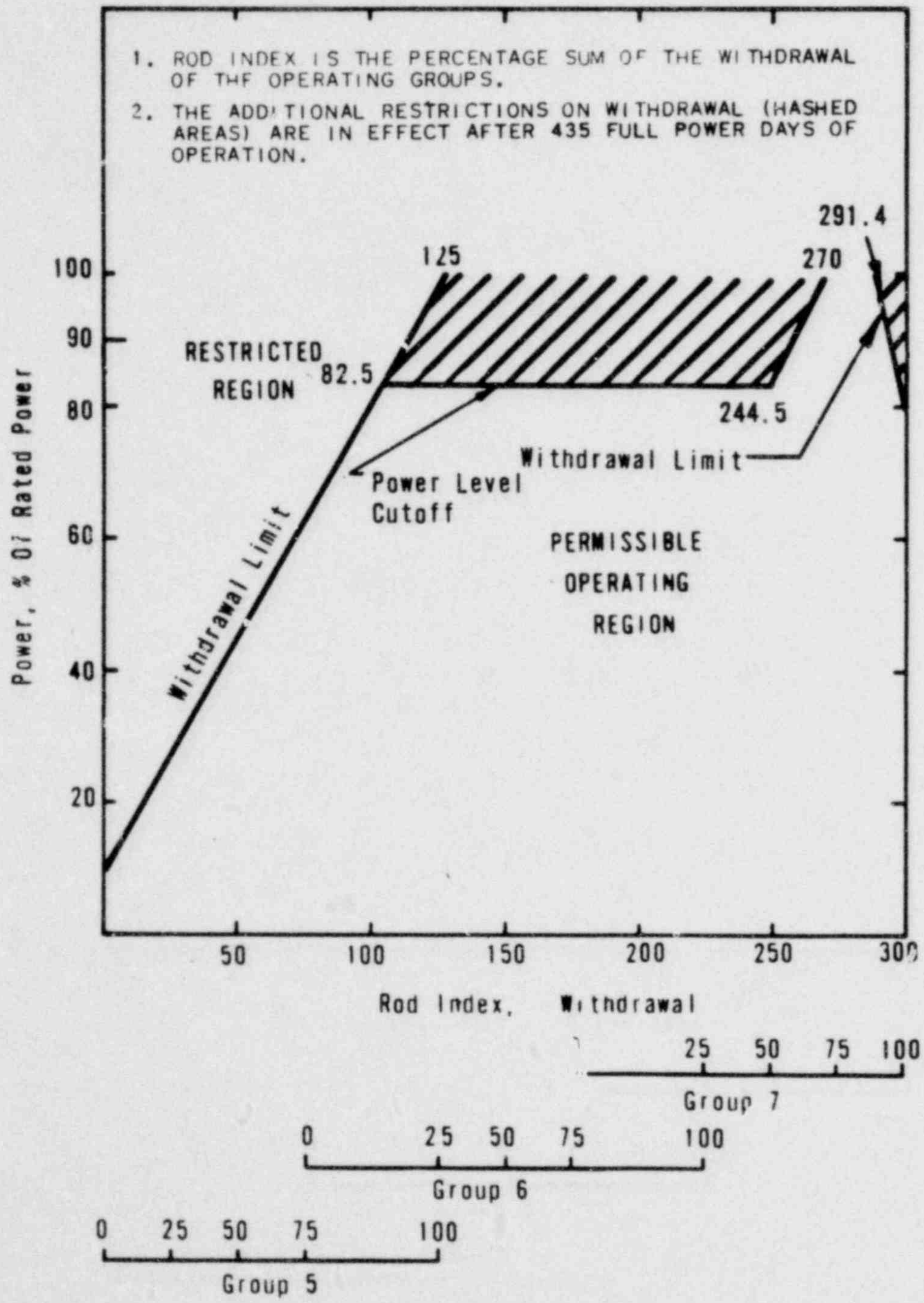
REFERENCES

[1]FSAR, Section 3.2.2.1.2

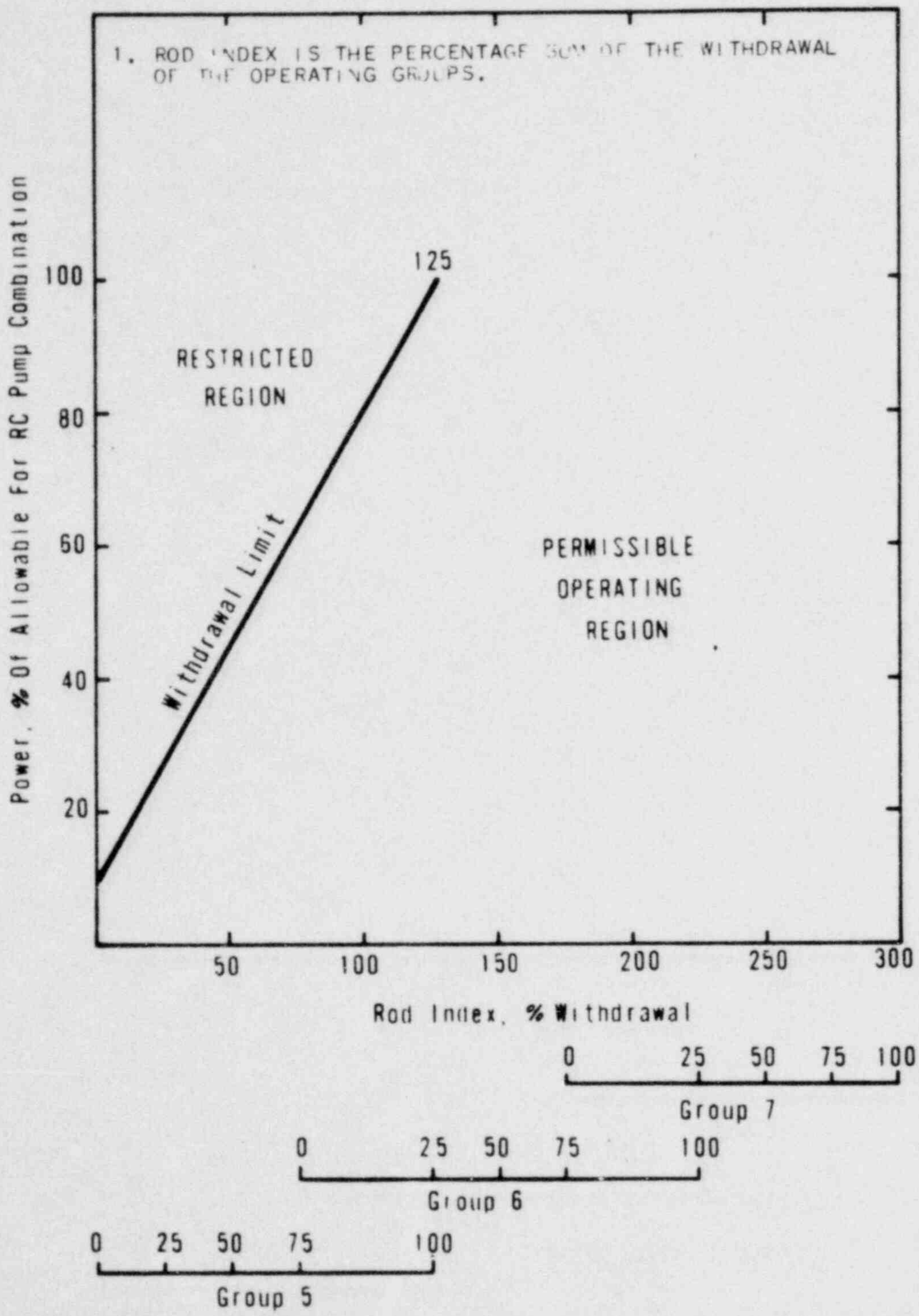
[2]FSAR, Section 14.2.2.2



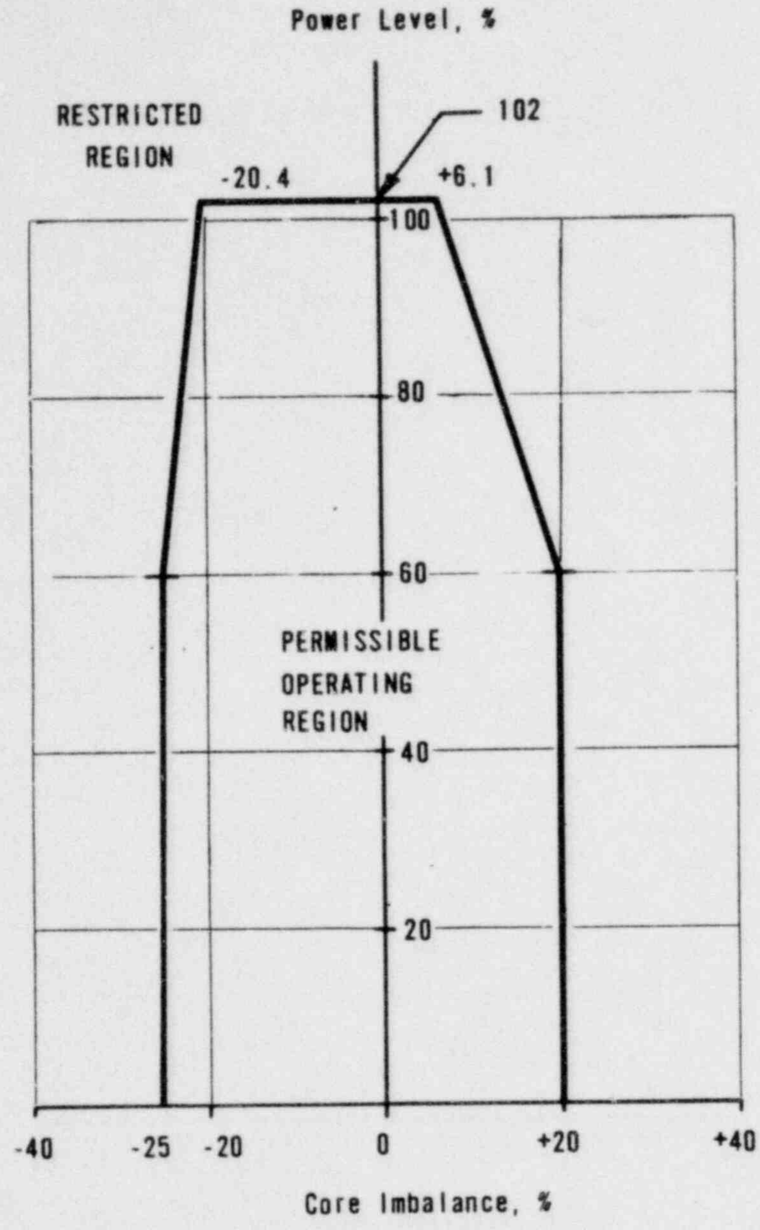
ARKANSAS POWER & LIGHT CO. ARKANSAS NUCLEAR ONE-UNIT 1	CONTROL ROD GROUP WITHDRAWAL LIMITS FOR 4 PUMP OPERATION	FIG. NO. 3.5.2-1A



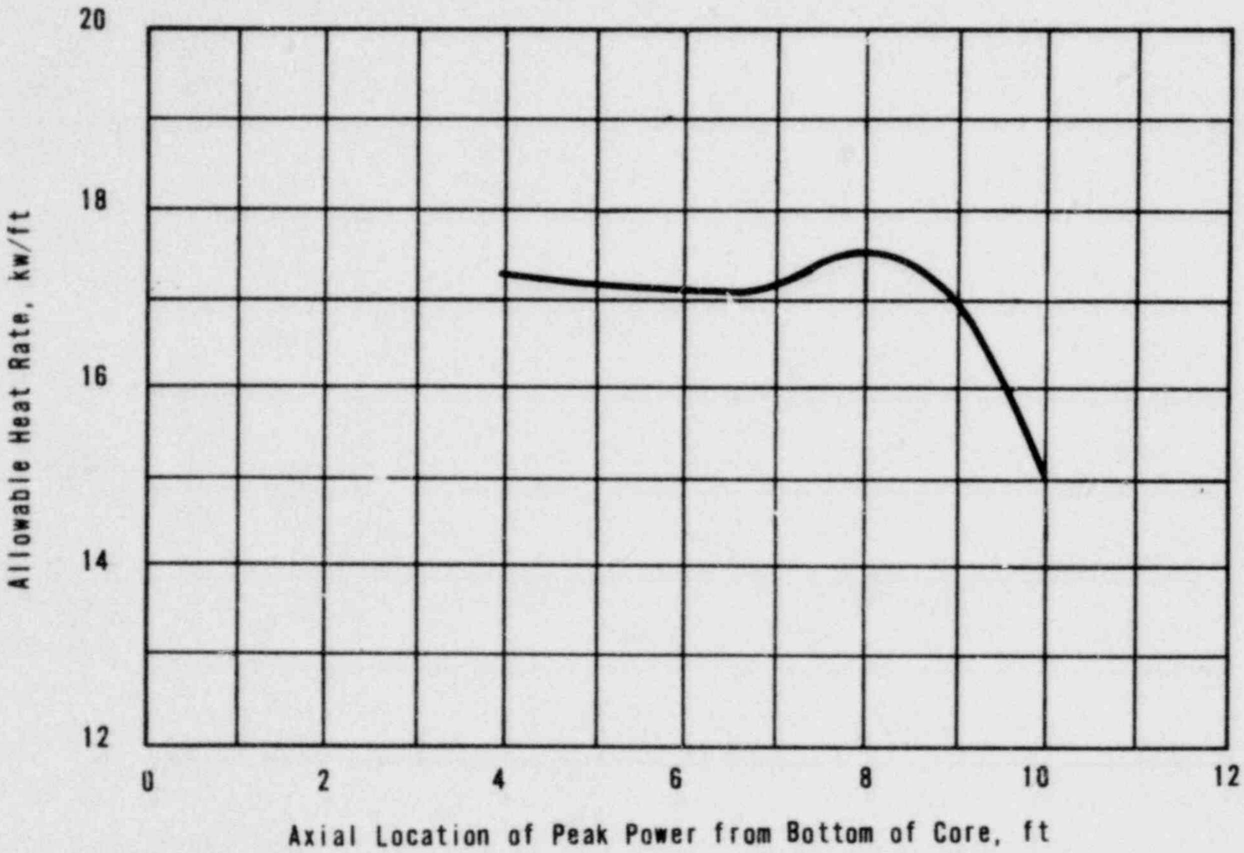
ARKANSAS POWER & LIGHT CO. ARKANSAS NUCLEAR ONE-UNIT 1	CONTROL ROD GROUP WITHDRAWAL LIMITS FOR 4 PUMP OPERATION	FIG. NO. 3.5.2-1B



ARKANSAS POWER & LIGHT CO. ARKANSAS NUCLEAR ONE-UNIT 1	CONTROL ROD GROUP WITHDRAWAL LIMITS FOR 3 AND 2 PUMP OPERATION	FIG.NO. 3.5.2-2



ARKANSAS POWER & LIGHT CO ARKANSAS NUCLEAR ONE-UNIT 1	OPERATIONAL POWER IMBALANCE ENVELOPE	FIG. NO. 3.5.2-3



ARKANSAS POWER & LIGHT CO. ARKANSAS NUCLEAR ONE-UNIT 1	LOCA LIMITED MAXIMUM ALLOWABLE LINEAR HEAT RATE	FIG. NO. 3.5.2-4

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	$\leq$30 psig (44.7 psia)
	High Pressure Injection	$\leq$ 4 psig (18.7 psia)
	Start of Reactor Building Cooling and Reactor Building Isolation	$\leq$ 4 psig (18.7 psia)
	Reactor Bldg. Ventilation	$\leq$ 4 psig (18.7 psia)
	Low Pressure Injection	$\leq$ 4 psig (18.7 psia)
	Penetration Room Ventilation	$\leq$ 4 psig (18.7 psia)
Low Reactor Coolant System Pressure **	High Pressure Injection	$\geq$ 1500 psig
	Low Pressure Injection	$\geq$ 1500 psig

*May be bypassed during reactor building leak rate test.

**May be bypassed below 1750 psig and is automatically reinstated above 1750 psig.

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 1500 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]

REFERENCE

(1) FSAR, Section 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 (Table 2.3-1) at least 23 individual incore detectors shall be operable to check gross core power distribution and 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, one 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 nuclear instrumentation calibration and for core power distribution verification.

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

1. Calibration of the split detectors at initial reactor start-up, during the power escalation program, and periodically 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 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 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. 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.

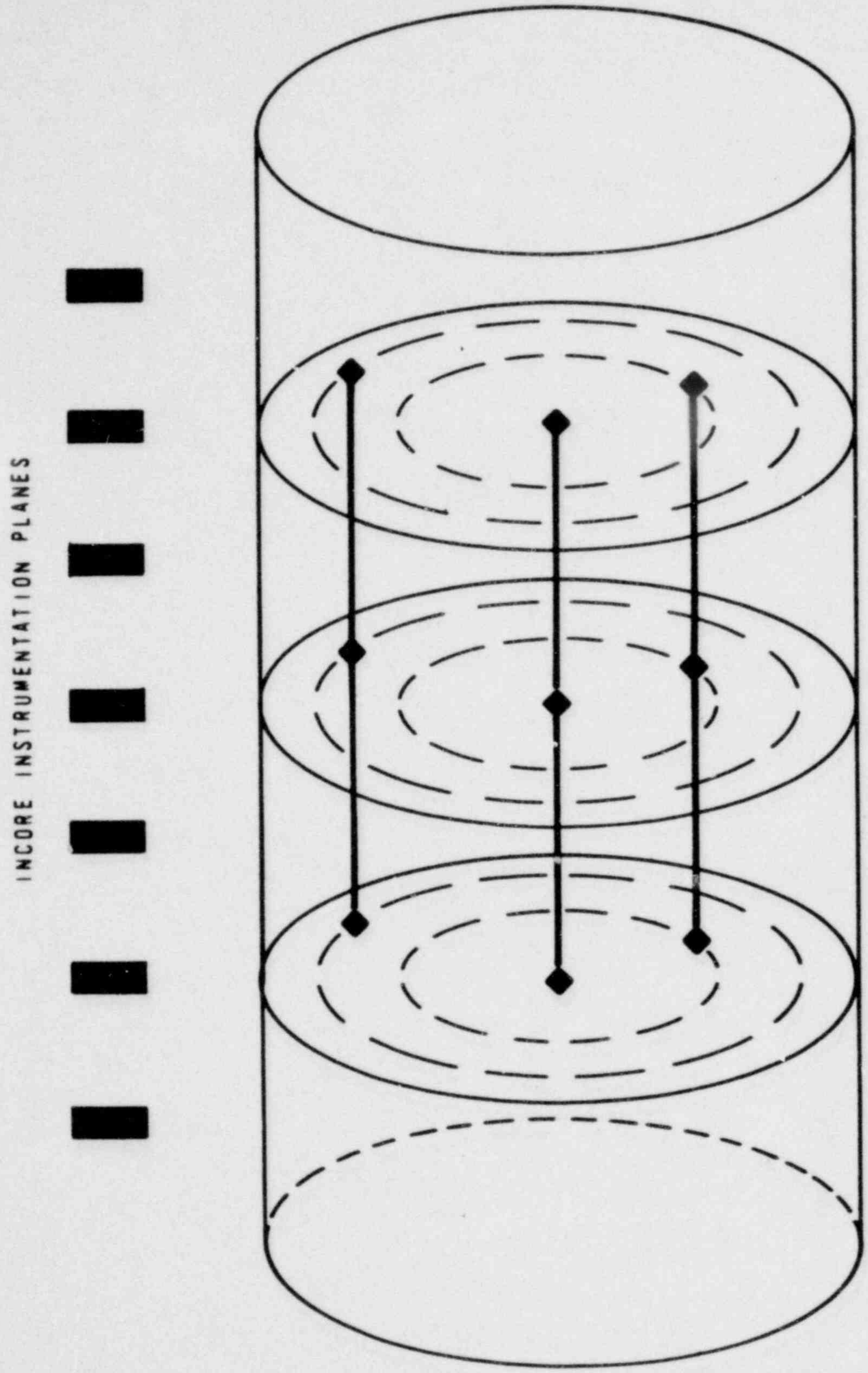
2. Figure 3.5.4-2 shows a typical detection scheme which will indicate the radial power distribution with 16 individual detectors. The readings from 2 detectors in a radial quadrant at either plane can be compared with readings from the other quadrants to measure a radial flux tilt.

3. 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.

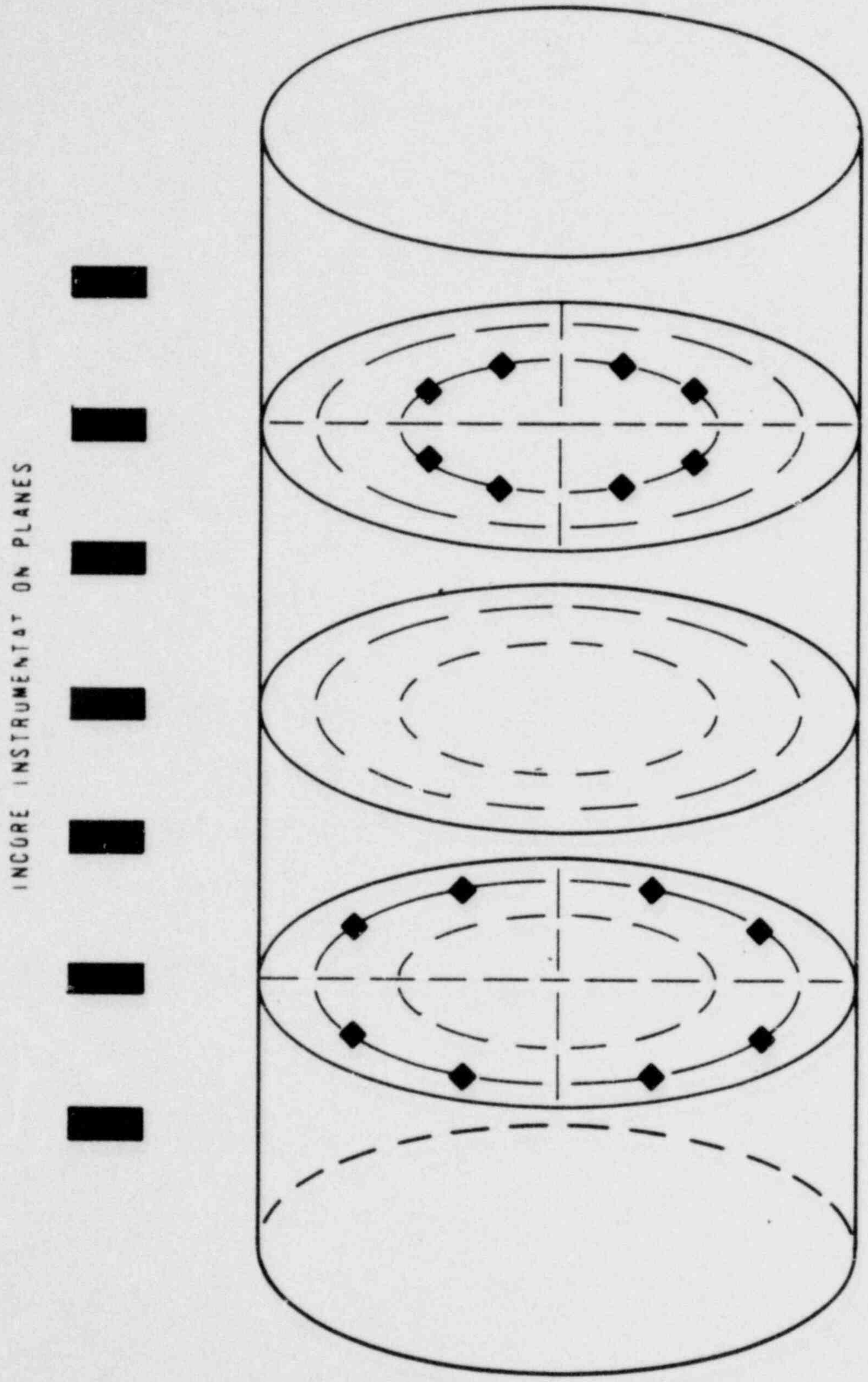
E. At least 23 specified incore detectors will be operable to check power distribution above 80 percent power determined by reactor coolant pump combination. These incore detectors will be read out either on the computer or on a recorder. If a set of 23 detectors in specified locations is not operable,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

REFERENCE

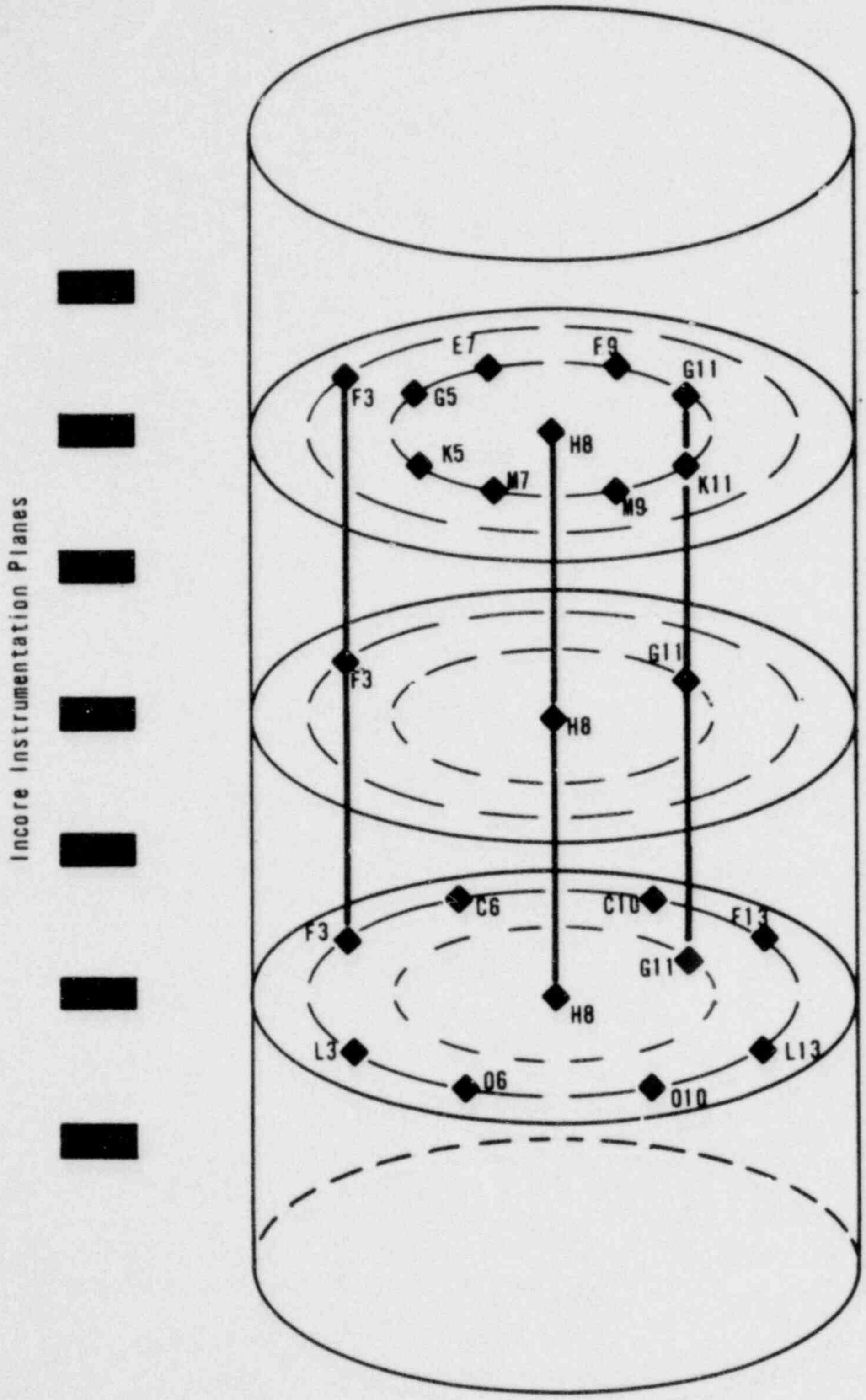
(1) FSAR, Section 4.1.1.3



ARKANSAS POWER & LIGHT CO. ARKANSAS NUCLEAR ONE-UNIT 1	INCORE INSTRUMENTATION SPECIFICATION AXIAL IMBALANCE INDICATION	FIG. NO. 3.5.4-1



ARKANSAS POWER & LIGHT CO. ARKANSAS NUCLEAR ONE-UNIT 1	INCORE INSTRUMENTATION SPECIFICATION RADIAL FLUX TILT INDICATION	FIG. NO. 3.5.4-2



ARKANSAS POWER & LIGHT CO. ARKANSAS NUCLEAR ONE-UNIT 1	INCORE INSTRUMENTATION SPECIFICATION	FIG. NO. 3.5.4-3

3.6 REACTCR BUILDING

Applicability

..pplies to the integrity of the reactor building.

Objective

To assure reactor building integrity.

Specification

3.6.1 Reactor building integrity shall be maintained whenever all three (3) 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 Reactor building integrity shall be maintained when the reactor coolant system is open to the reactor building 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% $\Delta k/k$ shall not be made by control rod motion or boron dilution whenever reactor building integrity is not in force.

3.6.4 The reactor shall not be taken critical or remain critical if the reactor building internal pressure exceeds 3.0 psig or a vacuum of 5.5 inches Hg.

3.6.5 Prior to criticality following a refueling shutdown, a check shall be made to confirm that all manual reactor building isolation valves which should be closed are closed and locked, as required.

3.6.6 If, while the reactor is critical, a reactor building isolation valve is determined to be inoperable in a position other than the closed position, the other reactor building isolation valve (except for check valves) 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 operable valve will be closed.

Bases

The reactor coolant system conditions of cold shutdown assure that no steam will be formed and hence there will be no pressure buildup in the reactor building 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 3.0 psi greater than the internal pressure. The design external pressure of 3.0 psi corresponds to a margin of 0.5 psi above the differential pressure that could be developed if the building is sealed with an internal temperature of 110°F and the building is subsequently cooled to an internal temperature of less than 50°F.

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

References

FSAR, Section 5

3.7 Auxiliary Electrical Systems

Applicability

Applies to the auxiliary electrical power systems.

Objectives

To specify conditions of operation for plant station power necessary to ensure safe reactor operation and combined availability of the engineered safety features.

Specifications

3.7.1 The reactor shall not be heated or maintained above 200°F unless the following conditions are met (except as permitted by Paragraph 3.7.2):

A. Any one of the following combinations of power sources operable:

1. Startup transformer No. 1 and Startup Transformer No. 2.

2. Startup transformer No. 2 and Unit Auxiliary Transformer provided that the latter one is connected to the 22kV line from the switchyard rather than to the generator bus.

B. All 4160 V switchgear, 480 V load centers and 480 V motor control centers in both of the ESAS distribution systems are operable and are being powered from either one of the two startup transformers or the unit auxiliary transformer.

C. Both diesel generator sets are operable and both diesel fuel oil storage tanks are full.

D. Both station batteries are operable and each is capable of supplying power to the 125V d-c distribution system. At least 2 of the 3 battery chargers are operable.

E. At least 2 of 3 d-c control power sources to the 125V d-c switchyard distribution system are operable.

3.7.2 A. The specifications in 3.7.1 may be modified to allow one of the following conditions to exist after the reactor has been heated above 200°F. Except as indicated in the following conditions, if any of these conditions are not met, a hot shutdown shall be initiated within 12 hours. If the condition is not cleared within 24 hours, the reactor shall be brought to cold shutdown within an additional 24 hours.

B. In the event that one of the offsite power sources specified in 3.7.1.A (1 or 2) is inoperable, reactor operation may continue for up to 24 hours if the availability of the diesel generators is immediately verified.

C. Either one of the two diesel generators may be inoperable for up to 7 days in any month provided that during such 7 days the operability of the remaining diesel generator is demonstrated immediately and daily thereafter, there are no inoperable ESF components associated with the operable diesel generator, and provided that the two sources of off-site power specified in 3.7.1.A(1) or 3.7.1.A(2) are available.

D. Any 4160V, 480V, or 120V switchgear, load center, motor control center, or distribution panel in one of the two ESF distribution systems may be inoperable for up to 8 hours, provided that the operability of the diesel generator associated with the operable ESF distribution system is demonstrated immediately and all of the components of the operable distribution system are operable. If the ESF distribution system is not returned to service at the end of the 8 hour period, specification 3.7.2.A shall apply.

E. Two station battery chargers may be inoperable for 8 hours, after which specification 3.7.2.A shall apply.

F. One of the two station batteries and the associated distribution system may be inoperable for 8 hours provided that there are no inoperable safety related components associated with the remaining station battery which are redundant to the inoperable station battery and the operability of the diesel generator is verified immediately. If the battery is not returned to service at the end of the 8 hour period, specification 3.7.2.A shall apply.

G. Two control power sources from the plant to the switchyard and the attendant distribution system may be inoperable for 8 hours, after which specification 3.7.2.A shall apply.

3.7.3 Any degradation beyond those conditions specified in 3.7.1 and 3.7.2 shall be cause to initiate an abnormal occurrence report per specifications 6.5 and 6.12.

3.8 FUEL LOADING AND REFUELING

Applicability

Applies to fuel loading and refueling operations.

Objective

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

Specification

3.8.1 Radiation levels in the reactor building refueling area shall be monitored by instrument RE-8017. Radiation levels in the spent fuel storage area shall be monitored by instrument RE-8009. 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 continuously monitored by at least two neutron flux monitors, each with continuous indication available, whenever core geometry is being changed. When core geometry is not deing changed, at least one neutron flux monitor shall be in service

3.8.3 At least one decay heat removal pump and cooler and its cooling water supply 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 closed.

3.8.8 When two irradiated fuel assemblies are being moved simultaneously by the bridges within the fuel transfer canal, a minimum of 10 feet separation shall be maintained between the assemblies at all times.

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 shall be tested and verified to be operable within 7 days 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 All fuel handling in the Auxiliary Building shall cease upon notification of the issuance of a tornado watch for Pope, Yell, Johnson, or Logan counties in Arkansas. Fuel handling operat[illegible] in progress will be completed to the extent necessary to place the fuel handling bridge and crane in their normal parked and locked position.

3.8.13 No loaded spent fuel shipping cask shall be carried above or into the Auxiliary Building equipment shaft unless atmospheric dispersion conditions are equal to or better than those produced by Pasquill type D stability accompanied by a wind velocity of 2 m/sec. In addition, the railroad spur door of the Turbine Building shall be closed and the fuel handling area ventilation system shall be in operation.

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 as described in Section 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 being 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 shutdown margin indicated in Specification 3.8.4 will keep the core subcritical, even with all control rods withdrawn from the core.(2) The boron concentration will be maintained above 1800 ppm. Although this concentration is sufficient to maintain the core $k_{eff} \leq 0.99$ if all the control rods were removed from the core, only a few control rods will be removed at any one time during fuel shuffling and replacement. The k_{eff} with all rods in the core and with refueling boron concentration is approximately 0.9. 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.

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.

Because of physical dimensions of the fuel bridges, it is physically impossible for fuel assemblies to be within 10 feet of each other while being handled.

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.[3]

REFERENCES

(1) FSAR, Section 9.5

(2) FSAR, Section 14.2.2.3

(3) FSAR, Section 14.2.2.3.3

3.9 RADIOACTIVE DISCHARGE

This specification has been replaced by specification 2.4 of the environmental technical specifications (Appendix B to the operating license).

 (Next page is 66)

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 iodine-131 activity in the secondary side of a steam generator shall not exceed .26 µCi/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 incident is considered. As stated in FSAR Table 14-16, 205,000 pounds of water is released to the atmosphere via the relief valves. An exclusion distance dose limit of 1.5 rem is used.

The whole body dose is negligible since any noble gases entering the secondary coolant system are continuously vented to the atmosphere by the condenser vacuum pumps. 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 system leak rate and technical specification limiting activity. One-tenth of the contained iodine is assumed to reach the exclusion distance, making allowance for plateout and retention in water droplets. I-131 is assumed to contribute 70% of the total thyroid dose based on the ratio of I-131 to the total iodine isotopes given in Table 11-5 of the FSAR.

The maximum permissible concentration of I-131 in the steam generator is as follows:

Dose (rem) = C x V x B x DCF x (0.1) X/Q = 1.5
C = Secondary coolant activity (Dose equivalent curies of I-131/m^3)
V = Secondary water volume released to atmosphere (120 m^3)
B = Breathing rate (3.47×10^{-4} m^3/sec)
X/Q = Ground level release dispersion factor (6.5×10^{-4} sec/m^3)
DCF = 1.48×10^6 rem/Ci
0.1 = Fraction of activity released

From the above the dose equivalent curies = .365
Max permissible concentration I-131 = (.7) (Dose Equivalent Curies/m^3)
Max permissible concentration I-131 = (.7) (.365) = .26 uCi/$_{cc}$

3.11 EMERGENCY COOLING POND

Applicability

Applies to the emergency cooling pond.

Objective

To assure the availability of a sufficient supply of cooling water inventory in the emergency cooling pond.

Specification

3.11.1 The reactor shall not be critical unless the water level in the emergency cooling pond is equal to or greater than elevation 344 feet 0 inches, corresponding to 3 feet pond depth.

Bases

The requirement of Specification 3.11.1 provides for sufficient water inventory in the emergency cooling pond to handle a DBA with a concurrent failure of the Dardanelle Reservoir. This minimum level takes into account (1) water loss from evaporation due to both heat load and climatological conditions, (2) pond bottom irregularities and (3) suction pipe level at the pond. The minimum level also corresponds to the effective depth used to determine pond response and assures the applicability of the calculational model for pond evaporation.

3.12 MISCELLANEOUS RADIOACTIVE MATERIALS SOURCES

Applicability

Applies to byproduct, source, and special nuclear radioactive material sources.

Objective

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

Specification

3.12.1 The source leakage test performed pursuant to Specification 4.14 shall be capable of detecting the presence of 0.005 µCi of radioactive material on the test sample. If the test reveals the presence of 0.005 µCi 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 µCi or less of beta and/or gamma emitting material or 10 µCi or less of alpha emitting material.

3.12.2 A complete inventory of licensed radioactive materials in possession shall be maintained current at all times.

4 SURVEILLANCE STANDARDS

Specified surveillance intervals may be adjusted plus or minus 25 percent to accommodate normal test and surveillance schedules. Surveillance requirements are not applicable when the plant operating conditions are below those requiring operability of the designated component. However, the required surveillance must be performed prior to reaching the operating conditions requiring operability. For example, instrumentation requiring twice per week surveillance when the reactor is critical need not have the required surveillance when the reactor is shutdown.

4.1 OPERATIONAL SAFETY ITEMS

Applicability

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

Objective

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

Specification

a. The minimum frequency and type of surveillance required for reactor protective system and engineered safeguards system instrumentation when the reactor is critical shall be as stated in Table 4.1-1.

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

c. Discrepancies noted during surveillance testing will be corrected and recorded.

d. A power distribution map shall be made to verify the expected power distribution at periodic intervals at least 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. Futhermore, 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 plant systems, when the plant is in operation, the minimum checking frequency stated is deemed adequate for reactor system instrumentation.

Calibration

Calibration shall be performed to assure the presentation and acquisition of accurate information. The nuclear flux (power range) channels shall be calibrated at least twice weekly (during steady state operating conditions) against a heat balance standard to compensate for instrumentation drift. During non-steady state operation, the nuclear flux channels shall be calibrated daily to compensate for instrumentation drift and changing rod patterns and core physics parameters.

Other channels are 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 for the nuclear flux (power range) channels, and once each refueling period for the process system channels is considered acceptable.

Testing

On-line testing of reactor protective channels is required once every 4 weeks on a rotational or staggered basis. The rotation scheme is designed to reduce the probability of an undetected failure existing within the system and to minimize the likelihood of the same systematic test errors being introduced into each redundant channel.

The rotation schedule for the reactor protective channels is as follows:

Channels A, B, C, D	Before Startup if shutdown greater than 24 hours.
Channel A	One Week After Startup
Channel B	Two Weeks After Startup
Channel C	Three Weeks After Startup
Channel D	Four Weeks After Startup

The reactor protective system instrumentation test cycle is continued with one channel's instrumentation tested each week. Upon detection of a failure that prevents trip action, all instrumentation associated with the protective channels will be tested after which the rotational test cycle is started again. If actuation of a safety channel occurs, assurance will be required that actuation was within the limiting safety system setting.

The protective channels coincidence logic and control rod drive trip breakers are trip tested every four weeks. The trip test checks all logic combinations and is to be performed on a rotational basis. The logic and breakers of the four protective channels shall be trip tested prior to startup and their individual channels trip tested on a cyclic basis. Discovery of a failure requires the testing of all channel logic and breakers, after which the trip test cycle is started again.

The equipment testing and system sampling frequencies specified in Table 4.1-2 and Table 4.3-3 is considered adequate to maintain the status of the equipment and systems to assure safe operation.

REFERENCE

FSAR Section 7.1.2.3.4

Table 4.1-1

Instrument Surveillance Requirements

Channel Description	Check	Test	Calibrate	Remarks
1. Protective Channel Coincidence Logic	NA	M	NA	
2. Control Rod Drive Trip Breaker	NA	M	NA	
3. Power Range Amplifier	NA	NA	T/W(1)	(1) Heat balance calibration twice weekly under steady state operating conditions, daily under non-steady state operating conditions.
4. Power Range Channel	S M(1)	M	M(1)(2)	(1) Using incore instrumentation. (2) Axial offset upper and lower chambers monthly and after each startup if not done previous week.
5. Intermediate Range Channel	S	P/M	NA	
6. Source Range Channel	S(1)	P	NA	(1) When in service.
7. Reactor Coolant Temperature Channel	S	M	R	
8. High Reactor Coolant Pressure Channel	S	M	R	
9. Low Reactor Coolant Pressure Channel	S	M	R	
10. Flux-Reactor Coolant Flow Comparator	S	M	R	
11. Reactor Coolant Pressure Temperature Comparator	S	M	R	
12. Pump Flux Comparator	S	M	R	

Table 4.1-1 (Cont'd)

Channel Description	Check	Test	Calibrate	Remarks
13. High Reactor Building Pressure Channel	S	M	R	
14. High Pressure Injection Logic Channel	NA	M	NA	
15. High Pressure Injection Analog Channels				
a. Reactor Coolant Pressure Channel	S	M (1)	R	(1) Including test of shutdown bypass function (ECCS bypass function).
b. Reactor Building 4 psig Channel	S	M	R	
16. Low Pressure Injection Logic Channel	NA	M	NA	
17. Low Pressure Injection Analog Channels				
a. Reactor Coolant Pressure Channel	S	M (1)	R	(1) Including test of shutdown bypass function (ECCS bypass function).
b. Reactor Building 4 psig Channel	S	M	R	
18. Reactor Building Emergency Cooling and Isolation System Logic Channel	NA	M	NA	
19. Reactor Building Emergency Cooling and Isolation System Analog Channels				
a. Reactor Building 4 psig Channels	S	M	R	

Table 4.1-1 (Cont'd)

Channel Description	Check	Test	Calibrate	Remarks
20. Reactor Building Spray System Logic Channels	NA	M (1)	NA	(1) Including RB spray pump, spray valve, and chem. add. valve logic channels.
21. Reactor Building Spray System Analog Channels				
a. Reactor Building Pressure Channels	NA	M	R	
22. Pressurizer Temperature Channels	S	NA	R	
23. Control Rod Absolute Position	S(1)	NA	R	(1) Compare with Relative Position Indicator.
24. Control Rod Relative Position	S(1)	NA	R	(1) Check with Absolute Position Indicator.
25. Core Flooding Tanks				
a. Pressure Channels	S	NA	R	
b. Level Channels	S	NA	R	
26. Pressurizer Level Channels	S	NA	R	
27. Makeup Tank Level Channels	D	NA	R	
28. Radiation Monitoring Systems	W	M(1)	Q(2)	(1) Check functioning of self-checking feature on each detector. (2) R for those detectors inaccessible during normal operation
29. High and Low Pressure Injection Systems: Flow Channels	NA	NA	R	

Table 4.1-1 (Cont'd)

Channel Description	Check	Test	Calibrate	Remarks
30. Decay Heat Removal System Isolation Valve Automatic Closure And Interlock System	S(1)(2)	M(1)(3)	R	(1) Includes RCS Pressure Analog Channel (2) Includes CFT Isolation Valve Position (3) Shall Also Be Tested During Refueling Shutdown Prior to Pressurization
31. Turbine Overspeed Trip Mechanism	N/A	R	N/A	
32. Steam Line Break Instrumentation And Control		(Later)		
33. Diesel Generator Protective Relaying, Starting Interlocks And Circuitry	M	Q	N/A	
34. Off-site Power Undervoltage And Protective Relaying Interlocks And Circuitry	W	R	R	
35. Borated Water Storage Tank Level Indicator	W	NA	R	
36. Boric Acid Mix Tank				
a. Level Channel	NA	NA	R	
b. Temperature Channel	M	NA	R	

Table 4.1-1 (cont'd)

Channel Description	Check	Test	Calibrate	Remarks
37. Boric Acid Addition Tank				
a. Level Channel	NA	NA	R	
b. Temperature Channel	M	NA	R	
38. Sodium Thiosulfate Tank Level Indicator	NA	NA	R	
39. Sodium Hydroxide Tank Level Indicator	NA	NA	R	
40. Incore Neutron Detectors	M(1)	NA	NA	(1) Check Functioning
41. Emergency Plant Radiation Instruments	M(1)	NA	R	(1) Battery Check
42. Deleted				
43. Strong Motion Acceleographs	Q(1)	NA	Q	(1) Battery Check
44. ESAS Manual Trip Functions				
a. Switches & Logic	NA	R	NA	
b. Logic	NA	M	NA	
45. Reactor Manual Trip	NA	P	NA	
46. Reactor Building Sump Level	NA	NA	R	

Note: S - Each Shift; T/W - Twice per Week; R - Each Refueling Period

D - Daily; B/M - Every 2 Months; NA - Not Applicable

W - Weekly; Q - Quarterly

M - Monthly; P - Prior to Each Startup if Not Done Previous Week

Table 4.1-2
Minimum Equipment Test Frequency

Item	Test	Frequency
1. Control Rods	Rod Drop Times of All Full Length Rods 1/	Each Refueling Shutdown
2. Control Rod Movement	Movement of Each Rod	Every Two Weeks Above Cold Shutdown Conditions
3. Pressurizer Code Safety Valves	Setpoint	One Within 2 Weeks Prior to or Following Each Refueling Shutdown
4. Main Steam Safety Valves	Setpoint	Four Within 2 Weeks Prior to or Following Each Refueling Shutdown
5. Refueling System Interlocks	Functioning	Start of Each Refueling Shutdown
6. Reactor Coolant System Leakage	Evaluate	Daily
7. Charcoal and High Efficiency Filters in Control Room, Penetration Room Ventilation System, Hydrogen Purge System, and Reactor Purge System	Charcoal and HEPA Filter for Iodine and Particulate Removal Efficiencies. DOP Test on HEPA Filters. Freon Test on Charcoal Filter Units 2/	Each Refueling Period and at Any Time Work on Filters Could Alter Their Integrity
8. Reactor Building Isolation Trip	Functioning	Each Refueling Shutdown
9. Service Water Systems	Functioning	Each Refueling Shutdown
10. Spent Fuel Cooling System	Functioning	Each Refueling Shutdown Prior to Use
11. Decay Heat Removal System Isolation Valve Automatic Closure and Isolation System	Functioning	Each Refueling Shutdown Prior to Repressurization

1/ Same as tests listed in section 4.7

2/ Same as tests listed in sections 4.4.3, 4.5.3, 4.11, and 4.12

Table 4.1-2 (Continued)
Minimum Equipment Test Frequency

	Item	Test	Frequency
12.	Flow Limiting Annulus on Main Feedwater Lines at Reactor Building Penetration	Verify, at normal operating conditions, that a gap of at least 0.025 inches exists between the pipe and the annulus.	One year, two years, three years, and every five years thereafter measured from date of initial test.

Table 4. 1-3

MINIMUM SAMPLING FREQUENCY

Item	Check	Frequency
1. Reactor Coolant	a. Gamma Isotopic Analysis	a. Monthly
	b. Radiochemical Analysis for Sr 89, 90	b. Monthly
	c. Tritium	c. Monthly
	d. Gross Beta & Gamma Activity (1)	d. 5 times/week
	e. Chemistry (Cl, F, and O_2)	e. 5 times/week
	f. Boron Concentration	f. 2 times/week
	g. Gross Alpha Activity	g. Monthly
	h. $\overline{E}$ Determination (2)	h. Semi-annually
2. Borated Water Storage Tank Water Sample	Boron Concentration	Weekly and after each makeup
3. Core Flooding Tank	Boron Concentration	Monthly and after each makeup
4. Spent Fuel Pool Water Sample	Boron Concentration	Monthly and after each makeup
5. Secondary Coolant	a. Gross Beta & Gamma Activity	a. Weekly
	b. Iodine Analysis (3)	
6. Sodium Hydroxide Tank	Sodium Hydroxide Concentration	Quarterly and after each makeup
7. Sodium Thiosulfate Tank	Sodium Thiosulfate Concentration	Quarterly and after each makeup

Table 4. 1-3

MINIMUM SAMPLING FREQUENCY

(1) When radioactivity level is greater than 10 percent of the limits of specification 3.1.4, the sampling frequency shall be increased to a minimum of once each day.

(2) $\bar{E}$ 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 radiochemical analysis for this purpose shall consist of a quantitative measurement of 95% of radionuclides in reactor coolant with half lives of 30 minutes. This is expected to consist of gamma isotopic analysis of dissolved and gaseous activities, radiochemical analysis for Sr 89, 90, and tritium analysis.

(3) When gross activity increases by a factor of two above background, an iodine analysis will be made and performed thereafter when the gross beta-gamma activity increases by 10 percent.

4.2 REACTOR COOLANT SYSTEM SURVEILLANCE

Applicability

Applies to the surveillance of the reactor coolant system pressure boundary.

Objective

To assure the continued integrity of the reactor coolant system pressure boundary.

Specification

4.2.1 Prior to initial unit operation, an ultrasonic test survey shall be made of reactor coolant system pressure boundary welds as required to establish preoperational integrity and base line data for future inspections.

4.2.2 Post operational inspections of components shall be made in accordance with the methods and intervals indicated in IS-242 and IS-261 of Section XI of the ASME Boiler and Pressure Vessel Code, 1971, including 1971 Winter addenda, except as follows:

IS-261 Item	Component	Exception
1.4	Primary Nozzle to Vessel Welds	1 RC inlet nozzle to be inspected after approx. 3 1/3 years operation. All four RC inlet nozzles to be inspected at or near the end of interval. Both RC outlet nozzles will be inspected after approx. 6 2/3 yrs. operation. One core flood nozzle will be inspected after 3 1/3 years operation and one core flood nozzle inspected near the end of interval
3.3	Safe Ends on Heat Exchanger	Not Applicable
4.1	Vessel Safe End Welds	Not Applicable
4.2	Valve Pressure Retaining Bolting Larger than 2"	Not Applicable
4.9	Integrally Welded Supports	Not Applicable
6.1	Valve Body Welds	Not Applicable
6.3	Valve to Safe End Welds	Not Applicable

IS-261 Item	Component	Exception
6.4	Bolting 2Ø	Not Applicable
6.6	Integrally Welded Valve Supports	Not Applicable

4.2.3 The structural integrity of the reactor coolant system boundary shall be maintained at the level required by the original acceptance standards throughout the life of the station. Any evidence, as a result of the tests outlined in Table IS-261 of Section XI of the code, that defects have developed or grown, shall be investigated.

4.2.4 To assure the structural integrity of the reactor internals throughout the life of the unit, the two sets of main internals bolts (connecting the core barrel to the core support shield and to the lower grid cylinder) shall remain in place and under tension. This will be verified by visual inspection to determine that the welded bolt locking caps remain in place. All locking caps will be inspected after hot functional testing and whenever the internals are removed from the vessel during a refueling or maintenance shutdown. The core barrel to core support shield caps will be inspected each refueling shutdown.

4.2.5 Sufficient records of each inspection shall be kept to allow comparison and evaluation of future inspections.

4.2.6 Complete surface and volumetric examination of the reactor coolant pump flywheels will be conducted coincident with refueling or maintenance shutdowns such that within a 10 year period after startup all four reactor coolant pump flywheels will be examined.

4.2.7 Vessel specimens will be withdrawn according to a schedule which may be modified to coincide with refueling or maintenance shutdowns. As a minimum requirement the withdrawal schedule will comply with ASTM-E-185-70.

Bases

The surveillance program has been developed to comply with Section XI of the ASME Boiler and Pressure Vessel Code Inservice Inspection of Nuclear Reactor Coolant Systems, 1971, including 1971 Winter Addenda edition.

The vessel specimen surveillance program will be based on equivalent exposure years which will assure that the first sample should be removed near the 50° NDTT shift point and the last sample should be removed near the end of the vessel design life.

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 2285 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 B 31.7, and ASME Boiler and Pressure Vessel Code, Section IX, 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 2285 psig (operating pressure +100 psi; ±50 psi is normal system pressure fluctuation),[1] it will be leak tight during normal operation.

REFERENCES

FSAR, Section 4

4.4 REACTOR BUILDING

4.4.1 Reactor Building Leakage Tests

Applicability

Applies to the reactor building.

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 Design Pressure Leakage Rate

The maximum allowable integrated leakage rate, L_a, from the reactor building at the 59 psig design pressure, P_p, shall not exceed 0.20 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, P_t, of 30 psig provided the resultant leakage rate, L_t, 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 design pressure and at the reduced pressure to be used in the periodic integrated leakage rat tests. The leakage rates thus measured shall be identified as L_{am} and L_{tm} respectively.

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

c. L_t shall not exceed $L_a \sqrt{\frac{P_t}{P_a}}$ for values of $\frac{L_{tm}}{L_{am}}$ above 0.7

d. If L_{tm}/L_{am} is less than 0.3, the initial integrated test results shall be subject to review by the AEC to establish an acceptable value of L_t.

Where (L_a) Design Basis Accident Leakage Rate at Pressure P_a

(L_t) Maximum Allowable Test Leakage Rate at Reduced Test Pressure P_t Under Test Condition

(L_{ao}) Maximum allowable operational leakage rate at pressure P_a

(L_{to}) Maximum allowable leakage rate at pressure P_t

(L_{am}) Initial Measured Leakage Rate at Pressure P_a

(L_{tm}) Initial Measured Leakage Rate at Pressure P_t

(P_a) Peak Test Pressure of 59 psig

(P_t) Reduced Test Pressure of 30 psig

4.4.1.1.3 Conduct of Tests

a. Leakage rate tests should not be started until essential temperature equilibrium has been attained. Containment test conditions should stabilize for a period of about four hours prior to the start of a leakage rate test.

b. The leakage rate test period shall extend to 24 hours of retained internal pressure. If it can be demonstrated to the satisfaction of those responsible for the acceptance of the containment structure that the leakage rate can be accurately determined during a shorter test period, the agreed upon shorter period may be used.

c. 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.

d. Closure of reactor building isolation valves for the purpose of the test shall be accomplished by the means provided for normal operation of the valves without preliminary exercised or adjustment.

4.4.1.1.4 Frequency of Test

After the initial preoperational leakage rate test, a set of three integrated rate tests shall be performed at approximately equal intervals during each 10 year service period, with the third test of each set coinciding with the end of each 10-year service period. The test may coincide with the plant inservice inspection shut down periods.

4.4.1.1.5 Conditions for Return to Criticality

If L_{tm} is less than L_{to}.
(L_{to} = 75% L_t)

or

If L_{am} is less than L_{ao}
(L_{ao} = 75% L_a)

4.4.1.1.6 Corrective Action Retest

If L_{tm} is greater than L_{to}, local leak tests will then be performed and the required repairs made. The integrated leakage test need not be repeated provided local measured leakage reduction achieved by repairs of individual leaks reduces the reactor building's overall measured leakage rate sufficiently such that L_{tm} is less than L_{to}.

4.4.1.1.7 Report of Test Results

The initial test report shall include a schematic arrangement of the leakage rate measurement system, the instrumentation used, the supplemental test method and the test program selected as applicable to the initial test and all subsequent periodic tests. The report shall contain an analysis and interpretation of the leakage rate test data to the extent necessary to demonstrate the acceptability of the reactor building's leakage rate in meeting the acceptance criteria.

4.4.1.2 Local Leakage Rate Tests

4.4.1.2.1 Scope of Testing

The local leak rate shall be measured for components in the following categories:

a. Reactor building penetrations whose design incorporates resilient seals, gaskets, or sealant compounds; piping penetrations fitte with expansion bellows.

b. Air lock door seals, including operating mechanism and penetrations with resilient seals which are part of the reactor building pressure boundary in the air lock structures.

c. Equipment and access doors with resilient seals or gaskets (seal-welded doors are excluded).

d. Components other than those listed in items a,b, and c above which develop leaks inservice and require repairs to meet the acceptance criterion of specification 4.4.1.1.5.

e. Reactor building isolation valves which provide a direct connection with the inside atmosphere of the reactor building.

f. Reactor building isolation valves which in the event of valve leakage on valve malfunction upon a reactor building isolation signal, may extend (outside of the reactor building) the boundary of the leakage-limiting barrier of the reactor primary containment beyond that included during the conduct of the tests required by specification 4.4.1.1 (includes instrument valves in lines connected to the reactor coolant pressure boundary)

g. Reactor building isolation valves in engineered safety systems penetrating the reactor building which, under post-accident conditions, are required to close following the termination of the safety function.

4.4.1.2.2 Conduct of Tests

a. Local leak rate tests shall be performed at a pressure of 59 psig.

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 tested penetrations and isolation valves shall not exceed 60% L_a.

4.4.1.2.4 Corrective Action

a. If at any time during operation it is determined that specification 4.4.1.2.3 is exceeded, repairs shall be initiated immediately

b. If conformance with specification 4.4.1.2.3 is not demonstrated within 72 hours following detection of excessive local leakage, the reactor shall be shutdown and placed in a condition such that reactor building integrity is not required. (Specification 3.6.1)

4.4.1.2.5 Test Frequency

Local leak detection tests shall be performed at a frequency of at least each refueling period, but in no case at intervals greater than two years except that:

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

(b) The personnel hatch and emergency hatch outer door seals shall be tested after each opening but no more frequently than daily during normal operation or weekly during refueling or cold shutdowns. In addition, a pressure test shall be performed on the personnel and emergency hatches every six months.

4.4.1.3 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 specified in 4.3.1.1 and 4.3.1.2 respectively.

4.4.1.4 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 each refueling period.

4.4.1.5 Visual Inspection

A visual examination of the accessible interior and exterior surfaces of the reactor building structure and its components shall be performed during each refueling shutdown and prior to any integrated leak test, to uncover any evidence of deterioration which may affect either the reactor building'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.

<u>Bases</u> (1)

The reactor building is designed for an internal pressure of 59 psig and a steam-air mixture temperature of 285 F. Prior to initial operation, the reactor building will be strength tested at 115% of design pressure and leak rate tested at the design pressure. The reactor building will also be leak tested prior to initial operation at not less than 50% of the design pressure. These tests will verify that the leakage rate from reactor building pressurization satifies the relationships given in the specification.

The performance of a periodic integrated leakage rate test during plant life provides a current assessment of potential leakage from the reactor building in case of an accident that would pressurize the interior of the reactor building. In order to provide a realistic appraisal of the integrity of the reactor building under accident conditions, the reactor building isolation valves are to be closed in the normal manner. The test pressure of 30 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 30 psig. The specification provides a relationship for relating the measured leakage of air at 30 psig to the potential leakage at 59 psig. 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 reactor building to a 0.20% leakage rate at 59 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 design pressure, of those portions of the reactor building envelope that are most likely to develop leaks during reactor operation (penetrations and isolation valves) and the low value of $.60L_a$ leakage that is specified as acceptable from tested penetrations and isolation valves. Third is the tendon stress surveillance program which provides assurance that an important part of the structural integrity of the reactor building is maintained.

References

(1) FSAR, Sections 5 and 13.

4.4.2 Structural Integrity

Applicability

Applies to the structural integrity of the reactor building.

Objective

To define the structural integrity of the reactor building.

Specification

4.4.2.1 Tendon Surveillance

For the tendon surveillance program, to be conducted over the life of the unit, twenty-one tendons shall be selected for periodic inspection for symptoms of material deterioration or force reduction. The surveillance tendons shall consist of ten hoop tendons, at least three in each of the three 240° sectors of the reactor building; five vertical tendons located at approximately equally spaced intervals; and six dome tendons, two in each of the three groups of dome tendons.

4.4.2.1.1 Lift-Off

Lift-off readings shall be taken for all 21 surveillance tendons.

4.4.2.1.2 Wire Inspection and Testing

A minimum of three surveillance tendons, one from each of the hoop, vertical, and dome families, shall be relaxed and one wire from each relaxed tendon shall be removed as a sample and visually inspected for corrosion or pitting. In addition, the applicable anchor assemblies shall be inspected for deleterious conditions, such as corrosion, cracks, missing wires and off size button heads. Tensile and elongation tests shall also be performed on a minimum of three specimens taken from the ends and middle of each of the wires. The specimens shall be the maximum length acceptable for the test apparatus to be used and shall include areas representative of significant corrosion or pitting.

After the wire removal, the tendons shall be retensioned to the stress level measured at the lift-off reading (and changes in shim thicknesses shall be recorded) and then checked by a final lift-off reading. The tendon elongation during retensioning shall be measured.

Should the inspection of one of the wires reveal any significant physical change (pitting or loss of area), additional wires shall be removed from the applicable surveillance tendons and inspected to determine the extent and cause change. The sheathing filler will be sampled and inspected for changes in physical appearance.

4.4.2.2 Inspection Intervals and Reports

The inspection intervals, measured from the date of the initial structural test, shall be one year, two years, three years, , and every five years thereafter or as modified based on experience. Tendon surveillance may be conducted during reactor operation provided design conditions regarding loss of adjacent tendons are satisfied at all times.

A quantitative analytical report covering results of each inspection shall be submitted (required by Technical Specification 6.7) and shall especially address the following conditions, should they develop:

(1) Broken wires.

(2) The force-time trend line for any tendon, when extrapolated, that extends beyond either the upper or lower bounds of the predicted design band.

(3) Unexpected changes in tendon conditions or sheathing filler properties.

4.4.2.3 End Anchorage Concrete Surveillance

a. The end anchorages of the surveillance tendons and adjacent concrete surface will be inspected.

b. The inspection interval will be one-half year and one year after the structural integrity test.

c. The selected inspection locations shall include:

(1) Four (4) locations on one buttress (hoop tendon anchorage)

(2) Two (2) locations on the top of the ring girder (vertical tendon anchorage).

(3) One (1) location on the ring girder (dome tendon anchorage).

d. The inspection of the selected anchorage area shall include documenting the areas by sketches. The sketches will include:

(1) The time of inspection

(2) The mapping of the predominant visible concrete crack patterns

(3) The measurement of the crack widths, length, orientation and location of cracks, by use of optical comparators or wire feeler gauges.

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

f. The acceptance criteria shall be as follows:

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

4.4.2.4 Liner Plate Surveillance

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

a. Two areas which have inward deformations relative to a short fixed chord. 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-inch centers. The measurements shall be accurate to ± .01 inch. Temperature readings shall be obtained on both the liner plate and outside reactor building 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 and marked.

4.4.2.4.2 Shortly after the initial pressure test and at the first scheduled refueling shutdown, reexamination of the areas identified in Section 4.4.2.4.1 shall be made. Measurements of the inward deformations and observations of any strain concentrations shall be made.

4.4.2.4.3 If the difference in the measured inward deformations, relative to the measurements derived immediately after the structural test is significant (for a particular location), and/or changes in strain concentration exist, an investigation shall be made. The investigation will determine any necessary corrective action.

4.4.2.4.4 The surveillance program shall be discontinued after the inspection made during the refueling shutdown if no corrective action was needed. If corrective action is required, the frequency of inspection for a continued surveillance program shall be determined.

Bases

Provisions have been made for an inservice surveillance program, covering 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, the following representative tendon groups have been selected for surveillance:

Hoop - Ten tendons, at least three in each of the three 240° sectors of the reactor building.

Vertical - Five tendons located at approximately equally spaced intervals.

Dome - Six tendons, two located in each of the three groups of dome tendons.

The inspection of at least one wire from three of the surveillance tendons is considered sufficient representation to detect the presence of any wide spread 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.

The liner plate surveillance is based on the requirement to monitor the liner plate performance as a membrane to preserve the required leak tightness of the reactor building.

4.4.3 Hydrogen Purge System

Applicability

Applies to testing the 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 period using written procedures. These tests shall consist of visual inspection, a flow measurement using flow instruments in the purging station and pressure drop measurements across the filter bank. Flow shall be design flow or higher, and pressure drops across the filter bank shall not exceed two times the pressure drop when new. Fan motors shall be operated continuously for at least one hour, and valves shall be proven operable. This test shall demonstrate that under simulated emergency conditions the system can be placed into operation as needed.

4.4.3.2 Filter Tests

During each refueling period, leakage tests using DOP on HEPA units and Freon-112 (or equivalent) on charcoal units shall be performed at design flow on the filter. Removal of 99.95% DOP by each entire HEPA filter unit and removal of 99.95% Freon-112 (or equivalent) by each entire charcoal absorber unit shall constitute acceptable performance. These tests must also be performed after any maintenance which may affect the strutural integrity of either the filtration system units or of the housing.

4.4.3.3 H_2 Detector Test

Hydrogen concentration instruments shall be calibrated each refueling period with proper consideration to moisture effect.

Bases

The purge system is composed of two purging stations. The purge system is operated as necessary to maintain the hydrogen concentration below the control limit. The exhaust from the purge system is discharged to the unit vent.

The purge rate is controlled through the use of a purging station consisting of two purge units. Each unit consists of a purge blower, dehumidifier, filter train, purge flowmeter, sample connection and flowmeter and associated piping and valves.

The blower is a rotary positive type. The dehumidifier consists of two redundant heating elements inserted in a section of ventilation duct. The function of the dehumidifier is to sufficiently increase the temperature of the entering air to assure 70 percent relative humidity entering the filter train with 100 percent saturated air entering the dehumidifier. The purpose of the dehumidifier is to assure optimum charcoal filter efficiency. Heating element control is provided by a thermoswitch. Humidity indication is provided downstream of the heating elements by a humidity readout gage. The filter train provides prefiltration, high efficiency particulate filtration and charcoal filtration. Face velocity to the charcoal filter is low. The charcoal filter is composed of a module consisting of two inch deep double tray carbon cells. Both the purge flow to the unit vent and the purge sample flow are metered using rotometers. Both of these rotometers have an accuracy of ± two percent of full scale, and each has remote readout capability. The purge sample activities can be collected, counted and analyzed in the radio-chemistry laboratory. Makeup air to the reactor building is supplied by fans using outside air.

Following a LOCA, there is adequate time before purging is required to permit checkout of the purging station.

References

FSAR Section 5.1.6

4.5 EMERGENCY CORE COOLING SYSTEM, REACTOR BUILDING COOLING SYSTEM PERIODIC TESTING AND PENETRATION ROOM VENTILATION SYSTEM

4.5.1 Emergency Core Cooling Systems

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 System Tests

4.5.1.1.1 High Pressure Injection System

(a) During each refueling period, a system test shall be conducted to demonstrate that the system is operable. A test signal will be applied to demonstrate actuation of the high pressure injection system for emergency core cooling operation.

(b) The test will be considered satisfactory if control board indication verifies that all components have responded to the actuation signal properly; all appropriate pump breakers shall have opened or closed and all valves shall have completed their travel.

4.5.1.1.2 Low Pressure Injection System

(a) During each refueling period, a system test shall be conducted to demonstrate that the system is operable. The test shall be performed in accordance with the procedure summarized below:

(1) A test signal will be applied to demonstrate actuation of the low pressure injection system for emergency core cooling operation.

(2) Verification of the engineered safeguard function of the service water system which supplies cooling water to the decay heat removal coolers shall be made to demonstrate operability of the coolers.

(b) The test will be considered satisfactory if control board indication verifies that all components have responded to the actuation signal properly; all appropriate pump breakers shall have opened or closed, and all valves shall have completed their travel.

4.5.1.1.3 Core Flooding System

(a) During each refueling period, a system test shall be conducted to demonstrate proper operation of the system. During this test, verification shall be made that the check valves in the core flooding tank discharge lines operate properly.

(b) The test will be considered satisfactory if control board indication of core flood tank level verifies that all check valves have opened.

4.5.1.2 Component Tests

4.5.1.2.1 Pumps

Approximately quarterly, the high pressure and low pressure injection pumps shall be started and operated to verify proper operation. Acceptable performance will be indicated if the pump starts, operates for fifteen minutes, and the discharge pressure and flow are within ±10% of the initial level of performance as determined using test flow paths.

4.5.1.2.2 Valves - Power Operated

(a) At intervals not to exceed three months each engineered safety feature valve in the emergency core cooling systems and each engineered safety feature valve associated with emergency core cooling in the service water system which are designed to open in the event of a LOCA shall be tested to verify operability.

(b) The acceptable performance of each power operated valve will be that motion is indicated upon actuation by appropriate signals.

Bases

The emergency core cooling systems are the principle reactor safety features 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 high pressure injection system under normal operating conditions has one pump operating. At least once per month, operation will be rotated to another high pressure injection pump. This will help verify that the high pressure injection pumps are operable.

The requirements of the service water system for cooling water are more severe during normal operation then under accident conditions. Rotation of the pump in operation on a monthly basis will verify that two pumps are operable.

The low pressure injection pumps are tested singularly for operability by opening the borated water storage tank outlet valves in the borated water storage tank fill line. This allows water to be pumped from the borated water storage tank through each of the injection lines and back to the tank.

With the reactor shutdown, 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 that the check valves have opened.

REFERENCE

FSAR Section 6

4.5.2 Reactor Building Cooling Systems

Applicability

Applies to testing of the reactor building cooling systems.

Objective

To verify that the reactor building cooling systems are operable.

Specification

4.5.2.1 System Tests

4.5.2.1.1 Reactor Building Spray System

(a) During each refueling period a system test shall be conducted to demonstrate proper operation of the system. A test signal will be applied to demonstrate actuation of the reactor building spray system (except for reactor building inlet valves to prevent water entering nozzles).

(b) Station compressed air or smoke will be introduced into the spray headers to verify the availability of the headers and spray nozzles at least every five years.

(c) The test will be considered satisfactory if visual observation and control board indication verifies that all components have responded to the actuation signal properly.

4.5.2.1.2 Reactor Building Cooling System

(a) During each refueling period, a system test shall be conducted to demonstrate proper operation of the system. The test shall be performed in accordance with the procedure summarized below:

(1) A test signal will be applied to actuate the reactor building cooling operation.

(2) Verification of the engineered safety features function of the service water system which supplies the reactor building coolers shall be made to demonstrate operability of the coolers.

(b) The test will be considered satisfactory if control board indication verifies that all components have responded to the actuation signal properly.

4.5.2.2 Component Tests

4.5.2.2.1 Pumps

At intervals not to exceed 3 months 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 fifteen minutes, and the discharge pressure and flow are within ±10% of a point on the pump head curve.

4.5.2.2.2 Valves

At intervals not to exceed three months each engineered safety features valve in the reactor building spray and reactor building cooling system and each engineered safety features valve associated with reactor building cooling in the service water system shall be tested to verify that it is operable.

Bases

The reactor building cooling system and reactor building spray system are designed to remove the heat in the reactor building atmosphere to prevent the building pressure from exceeding the design pressure.

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 closed, the reactor building spray injection valves can each be opened and closed by operator action. With the reactor building spray inlet valves closed, low pressure air or smoke can be blown through the test connections of the reactor building spray nozzles to demonstrate that the flow paths are open.

The equipment, piping, valves, and instrumentation of the reactor building 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 service water piping and valves outside the reactor building are inspectable at all times. Operational tests and inspections will be performed prior to initial startup.

Two service water pumps are normally operating. At least once per month operation of one pump is shifted to the third pump, so testing will be unnecessary.

The reactor building fans are normally operating, so testing is unnecessary.

Reference

FSAR, Section 6

4.5.3 Penetration Room Ventilation System

Applicability

Applies to testing of the reactor building penetration room ventilation system.

Objective

To verify that the penetration room ventilation system is operable.

Specification

4.5.3.1 System Tests

4.5.3.1.1 During each refueling period a system test shall be conducted to demonstrate proper operation of the system. This test shall consist of visual inspection, a flow measurement using the flow instrument installed at the outlet of each unit and pressure drop measurements across each filter unit. In addition, a test signal will be applied to demonstrate proper actuation of the penetration room ventilation system. Fan motors shall be operated continuously for at least one hour, and the louvers and other mechanical systems shall be proven operable and adjustable from their remote location.

4.5.3.1.2 The test will be considered satisfactory if control board indication verifies that all components have responded properly to the actuation signal, if flow rate through the system is design flow or higher, and if pressure drops across any filter bank do not exceed two times the pressure drop which existed when the filters were new.

4.5.3.2 Filter Tests

No less frequently than each normal refueling period, "in-place" leakage tests using DOP on HEPA units and Freon-112 (or equivalent) on charcoal units shall be performed at design flow on each filter train. Removal of 99.95% DOP by each entire HEPA filter unit and removal of 99.95% Freon-112 (or equivalent) by each entire charcoal adsorber unit shall constitute acceptable performance. These tests must also be performed after any maintenance which may affect the structural integrity of the filtration system units.

Bases

The penetration room ventilation system is designed to collect and process potential reactor building penetration room leakage to minimize environmental activity levels resulting from post accident reactor building leaks. The system consists of a sealed penetration room, two redundant filter trains, and two redundant fans discharging to the unit vent. The entire system is activated by a reactor building pressure engineered safety features signal and initially requires no operator action.

Each filter train is constructed with a prefilter, an absolute filter, and a charcoal filter in series. The design flow rate through each of these filters is 2000 scfm, which is significantly higher than the 1.25 scfm maximum leakage rate from the reactor building at a leak rate of 0.1% per day. Except for periodic ventilation of the penetration room, the penetration room ventilation system is not normally used. Refueling period testing of this system will show that the system is available for its engineered safety features function. During this test, the system will be inspected for such things as water, oil, or other foreign material,gasket deterioration in the HEPA units, and unusual or excessive noise or vibration when the fan motor is running.

Less frequent testing will verify the efficiency of the absolute and charcoal filters.

4.6 AUXILIARY ELECTRICAL SYSTEM TESTS

Applicability

Applies to the periodic testing and surveillance requirements of the auxiliary electrical system to ensure it will respond promptly and properly when required.

Specification

4.6.1 Diesel Generators

1. Each diesel generator shall be manually started each month and demonstrated to be ready for loading within 15 seconds. The signal initiating the start of the diesel shall be varied from one test to another (start with handswitch at control room panel and at diesel local control panel) to verify all starting circuits are operable. The generator shall be synchronized from the control room and loaded to full rated load and allowed to run until diesel generator operating temperatures have stabilized.

2. A test shall be conducted during each refueling outage to demonstrate that the emergency power system is available to carry load within 15 seconds of a simulated ES signal of the safety features system coincident with the loss of offsite power. The diesel generator shall be fully loaded and run for one hour after operating temperatures have stabilized.

3. Each diesel generator shall be given an inspection at least every refueling outage following the manufacture's recommendations for this class of standby service. The above tests will be considered satisfactory if all applicable equipment operates as designed.

4. During the monthly diesel generator test specified in Paragraph 1 above, the diesel starting air compressors shall be checked for operation and their ability to recharge the air receivers.

   Also monthly, the diesel oil transfer pumps shall be checked for operation and their ability to transfer oil to the day tank.

5. During each refueling outage, the capability of each starting air compressor to charge the air compressor to charge the air receivers from 0 to 225 psig within 2 hours shall be verified.

   Also at each refueling outage, the capacity of each diesel oil transfer pump shall be verified to be at least 10 gpm.

4.6.2 Station Batteries and Switchyard Batteries

1. The voltage, temperature and specific gravity of a pilot cell in each bank and the overall battery voltage of each bank shall be measured and recorded daily.

2. Measurements shall be made quarterly of voltage of each cell to the nearest 0.01 volt of the specific gravity of each cell, and of the temperature of every fifth cell in each bank. The level of the electrolyte shall be checked and adjusted as required. All data, including the amount of water added to any cell, shall be recorded.

3. During each refueling outage, a performance discharge test shall be conducted in accordance with the manufacturer's instructions, for the purpose of determining battery capacity.

4. Every quarter, the third battery charger, which is capable of being connected to either of the two 125V d-c distribution systems, shall be tested and loaded while connected to each bus for 30 minutes.

4.6.3 Emergency Lighting

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

Bases

The emergency power system provides power requirements for the engineered safety features in the event of a DBA Each of the two diesel generators is capable of supplying minimum required engineered safety features from independent buses. This redundancy is a factor in establishing testing intervals. The monthly tests specified above will demonstrate operability and load capacity of the diesel generator. The fuel supply and diesel starter motor air pressure are continuously monitored and alarmed for abnormal conditions. Starting on complete loss of off-site power will be verified by simulated loss-of-power tests at intervals not to exceed each refueling shutdown period.

Considering system redundancy, the specified testing intervals for the station batteries should be adequate to detect and correct any malfunction before it can result in system malfunction. Batteries will deteriorate with time, but precipitous failure is extremely unlikely. The surveillance specified is that which has been demonstrated over the years to provide an indication of a cell becoming unserviceable long before it fails.

Routine battery maintenance specified by the manufacturer includes regularly scheduled equalizing charges in order to retain the capacity of the battery. A test discharge should be conducted to ascertain the capability of the battery to perform its design function under postulated accident condition. An excessive drop of voltage with respect to time is indicative of required battery maintenance or replacement.

Testing of the emergency lighting is scheduled annually and is subject to review and modification if experience demonstrates a more effective test schedule.

References

FSAR, Section 8

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 inches travel) shall not exceed 1.66 seconds at reactor coolant full flow conditions or 1.40 seconds for 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 (9) 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 inches 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 (2) inches of travel every two (2) 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.

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 (9) inches. Conditions for operation with an inoperable rod are specified in Technical Specification 3.5.2.

REFERENCES

(1) FSAR, Section 14

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) is operating in its programmed functional position and group (rods 1 through 12, group 1-8).

Specification

4.7.2.1 Whenever the control rod drive patch panel is locked (after test, reprogramming, or maintenance) each control rod drive mechanism shall be selected from the control room and exercised by a movement of two inches or less to verify that the proper rod has responded as shown on the unit computer printout 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 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, (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 (b) the proper meter in the control room display bank in both absolute and relative meter positions. This type of comparative check will not assure detection of improperly connected cables inside the reactor building. For these, (Specification 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.

4.8 EMERGENCY FEEDWATER PUMP

Applicability

Applies to the periodic testing of the turbine and electric motor driven emergency feedwater pumps.

Objective

To verify that the emergency feedwater pump and associated valves are operable.

Specification

4.8.1 Test

1. The turbine and electric motor driven emergency feedwater pumps shall be operated every three months for a minimum of one hour.
2. The emergency feedwater valves shall be cycled every three months.
3. During each normal refueling shutdown, a functional test of the emergency feedwater system shall be made using the electric motor driven emergency feedwater pump.

4.8.2 Acceptance Criteria

This test shall be considered satisfactory if control board indication and visual observation of the equipment demonstrates that all components have operated properly.

Bases

The three (3) month testing frequency will be sufficient to verify that both emergency 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 cycling of the emergency valves will be done coincident with the pump testing, but not concurrently so that cold emergency feedwater is not pumped to the steam generator.

The functional test during normal refueling shutdown will verify that the flow path to the steam generators is open and that water reaches the steam generators from the emergency feedwater system. The test is done during shutdown to avoid thermal cycle to the emergency feedwater nozzles on the steam generator due to the lower temperature of the emergency feedwater.

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 periodically compared with the predicted value. If the difference between the observed and predicted steady-state concentrations reaches the equivalent of one percent in reactivity, an evaluation of this abnormal occurrance will be made to determine the cause of the discrepancy.

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 slop 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 than 1 percent $\Delta k/k$ would be unexpected, and its occurrence would be thoroughly investigated and evaluated.

The value of 1 percent $\Delta k/k$ is considered a safe limit since a shutdown margin of at least 1 percent $\Delta k/k$ with the most reactive rod in the fully withdrawn position is always maintained.

4.10 ENVIRONMENTAL SURVEILLANCE

This specification is now covered by the Environmental Technical Specification (Appendix B to the Operating License).

4.11 CONTROL ROOM EMERGENCY VENTILATION

Applicability

Applies to control room emergency ventilation system components.

Objective

To verify that the system and components will perform their design functions.

Specification

4.11.1 Operating Tests

System tests shall be performed at approximately quarterly intervals. These tests shall consist of visual inpection, a flow measurement using a flow instrument installed at the outlet of the unit and pressure drop measurements across each filter bank. Pressure drop across the filters shall not exceed twice that when they are clean. Fan motors shall be operated continuously for at least one hour and all dampers and other mechanical and isolation systems shall be proven operable.

4.11.2 Filter Tests

During each refueling period, "in-place" leakage tests using DOP on HEPA units and Freon-112 (or equivalent) on charcoal units shall be performed at design flow on each filter train. Removal of 99.95% DOP by each entire HEPA filter unit and removal of 99.95% Freon-112 (or equivalent) by each entire 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 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 one 100 percent capacity filter train which consists of a prefilter, high efficiency particulate filters, charcoal filters and a fan.

Since the system is not normally operated, a periodic test is required to insure 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 motor is running.

Annual testing will verify the efficiency of the charcoal and absolute filters.

4.12 REACTOR BUILDING PURGE SYSTEM

Applicability

Applies to testing reactor building purge exhaust unit.

Objective

To verify that the reactor building purge filters will perform their design function.

Specification

During each refueling period, leakage tests using DOP on the HEPA filter and Freon-112 (or equivalent) on the charcoal unit shall be performed. Removal of 99.95% DOP by the HEPA filter unit and removal of 99.95% 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 the filtration units or of the housing.

Bases

The reactor building purge exhaust unit is constructed with a prefilter, an absolute filter and a charcoal filter in series. This test will verify the efficiency of the absolute and charcoal filters.

4.13 EMERGENCY COOLING POND

Applicability

Applies to the emergency cooling pond.

Objective

To verify the availability of a sufficient supply of cooling water inventory in the emergency cooling pond.

Specification

4.13.1 The emergency cooling pond water level shall be recorded daily to ensure that the water level is equal to or greater than elevation 344 feet 0 inches, corresponding to 3 feet pond depth.

4.13.2 Soundings shall be made annually of the emergency cooling pond bottom to ensure that the required volume of water is available. Specifications 3.11.1 and 4.13.1 shall be modified as necessary to accommodate changes in bottom elevation.

Bases

The requirements of Specification 4.13 provide for verification of a sufficient water inventory in the emergency cooling pond to handle a DBA with a concurrent failure of the Dardanelle Reservoir. This specification ensures that specification 3.11.1 is met.

4.14 RADIOACTIVE MATERIALS SOURCES SURVEILLANCE

Applicability

Applies to leakage testing of byproduct, source, and special nuclear radioactive material sources.

Objective

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

Specification

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:

1. 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.

2. The periodic leak test required does not apply to sealed sources that are stored and not being used. The sources excepted from this test shall be tested for leakage prior to any use of transfer to another user unless they 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.

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

4.15 AUGMENTED INSERVICE INSPECTION PROGRAM FOR HIGH ENERGY LINES OUTSIDE OF CONTAINMENT

Applicability

Applies to welds in piping 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 Arkansas Nuclear One-Unit 1 this specification applies to six welds in the main steam and main feedwater lines identified as welds 6, 7, 23, 24, 55 and 56 on Figures A-7, A-8 and A-15 of the Final Safety Analysis Report.

Objective

To provide assurance of the continued integrity of the piping systems over their service lifetime.

Specifications

4.15.1 At the first refueling outage period, a volumetric examination shall be performed with 100 percent inspection of each weld in accordance with the requirements of ASME Code Section XI, Inservice Inspection of Nuclear Power Plant Components, to establish system integrity and baseline data.

4.15.2 The inservice inspection at each weld shall be performed in accordance with the requirements of ASME Code Section XI, Inservice Inspection of Nuclear Power Plant Components, with the following schedule:

(The inspection intervals identified below sequentially follow the baseline examination of 4.15.1).

First Inspection Interval

a. First 3-1/3 years (or nearest refueling outage)	100% volumetric inspection of each weld
b. Second 3-1/3 years (or nearest refueling outage)	100% volumetric inspection of each weld
c. Third 3-1/3 years (or nearest refueling outage)	100% volumetric inspection of each weld

Successive Inspection Intervals

Every 10 years thereafter (or nearest refueling outage)	Volumetric inspection of two of the welds at the expiration of each 1/3 of the inspection interval with a cumulative 100% 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.

4.15.3 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.

4.15.4 Examinations that reveal unacceptable structural defects in a weld during an inspection under 4.15.2 should 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.

4.15.5 Repairs, reexamination and piping pressure tests shall be conducted in accordance with Section XI of the ASME Code.

4.16 SPECIAL SURVEILLANCE

Applicability

Applies to miscellaneous surveillance items not covered by other specifications.

Objective

To provide special surveillance for items not covered by other specifications.

Specification

4.16.1 All hydraulic shock suppressors installed on safety related systems shall be inspected for proper operation during each refueling shutdown.

Bases

Experiences at other nuclear power plants have uncovered problems with the seal material and loss of hydraulic fluid of hydraulic shock suppressors. This problem is of a generic nature so the surveillance provided will ensure that the suppressors installed at Arkansas Nuclear One (which are of different manufacture than those which failed) do not experience the same failures.

5 DESIGN FEATURES

Specifications for design features are intended to cover characteristics of importance to each of the physical barriers, and to the maintenance of safety margins in the design.

5.1 SITE

Applicability

Applies to the location and extent of the exclusion area.

Objective

To define the location and the size of the site area as pertains to safety.

Specification

Arkansas Nuclear One-Unit 1 is located on a site consisting of approximately 1100 acres which provides for 0.65 statute mile exclusion radius from the reactor building. This exclusion area includes certain portions of the bed and banks of the Dardanelle Reservoir which are owned by the Federal Government. An easement authorizes AP&L to exclude all persons from these areas during periods of emergency. The site is approximately 6 statute miles WNW from the City of Russellville (Latitude 35°-18'-36" N, Longitude 93°-13'-53"W) in an area characterized by remoteness from population centers.

REFERENCES

FSAR, Section 2.2

5.2 REACTOR BUILDING

Applicability

Applies to those design features of the reactor building relating to operational and public safety.

Objective

To define the significant design features of the reactor building structure, reactor building isolation system, and penetration room ventilation system.

Specification

5.2.1 Reactor Building Structure

The reactor building completely encloses the reactor and the associated reactor coolant system. It is a fully continuous 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 tensioning system consisting of horizontal and vertical tendons. The dome has a three-way post tensioning system. The foundation slab is conventionally reinforced with high strength reinforcing steel. The entire structure is lined with 1/4" welded steel plate to provide vapor tightness.

The internal volume of the reactor building is approximately 1.91×10^6 cu. ft. The approximate inside dimensions are: diameter--116'; height--207'. The approximate thickness of the concrete forming the buildings are: cylindrical wall--3 3/4'; dome--3 1/4'; and the foundation slab--9'.

The concrete reactor building structure provides adequate 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 3.0 psi greater than the internal pressure. This corresponds to a margin of 0.5 psi above the differential pressure that could be developed if the building is sealed with an internal temperature of 110 F and it is subsequently cooled to an internal temperature of less than 50 F. Since the building is designed for this pressure differential, vacuum breakers are not required.

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, engineered safety features, and the combined influence of energy sources and heat sinks. (1)

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)

5.2.3 Penetration Room Ventilation System

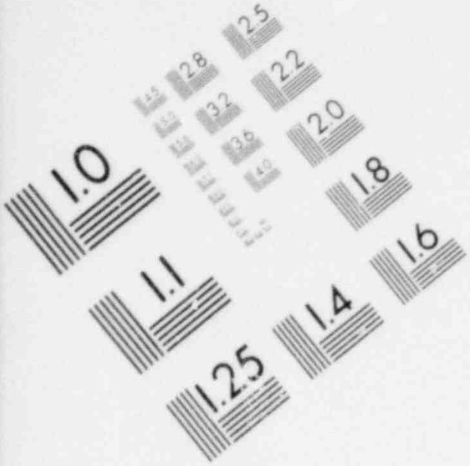
This system is designed to collect, control, and minimize the release of radioactive material from the reactor building to the environment in post-accident conditions. It may also operate intermittently during normal conditions as required to maintain satisfactory temperature in the penetrations rooms. When the system is in operation, a slightly negative pressure will be maintained in the penetration room to assure inleakage. (3)

References

(1) FSAR Section 5.1

(2) FSAR Section 5.1.5

(3) FSAR Section 6.5



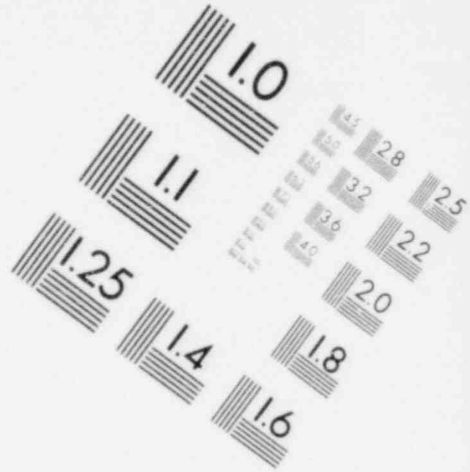
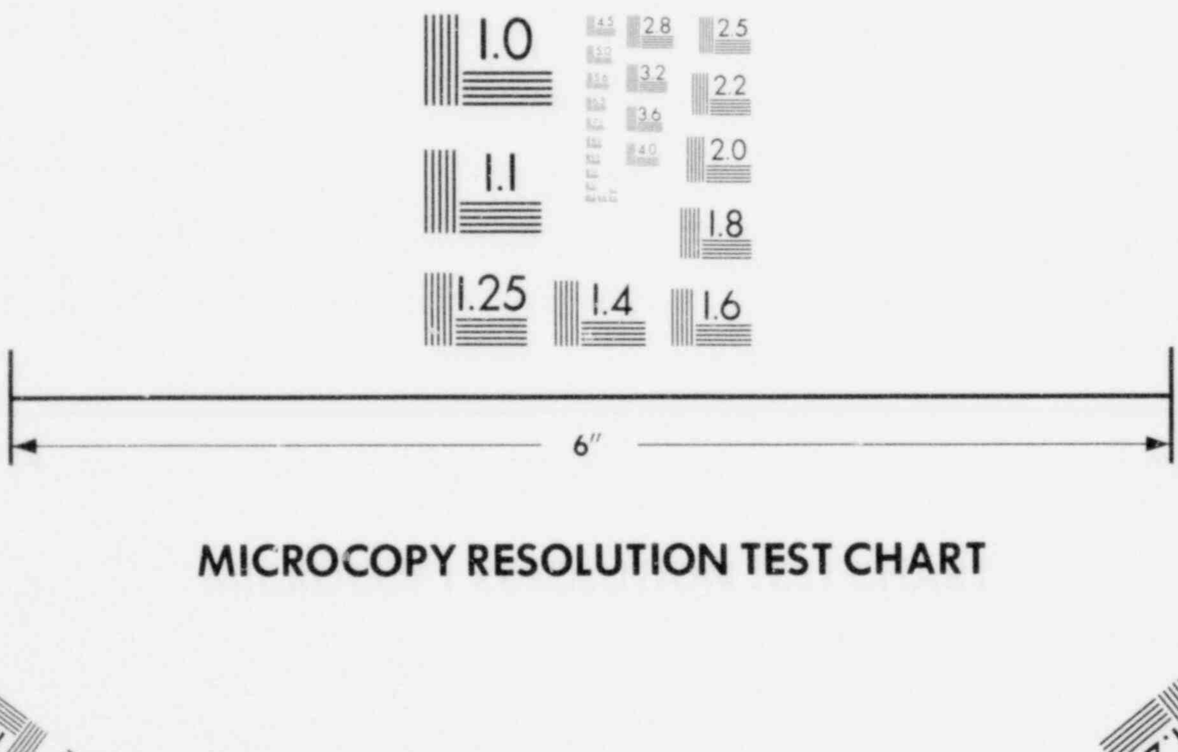
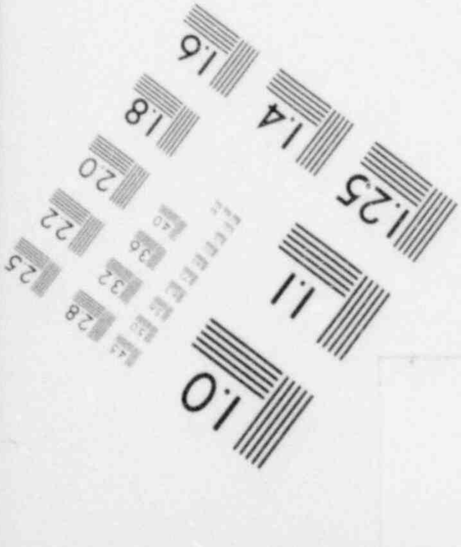
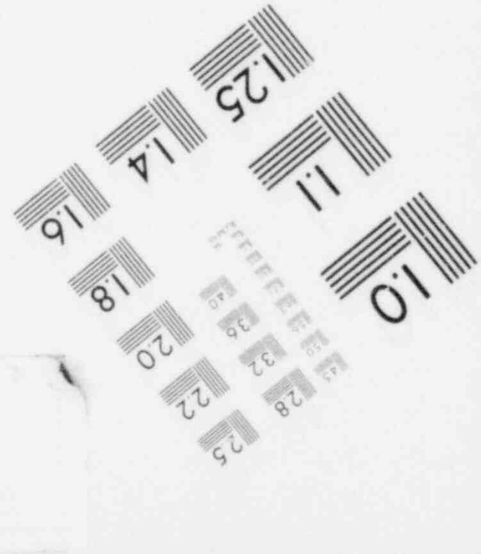


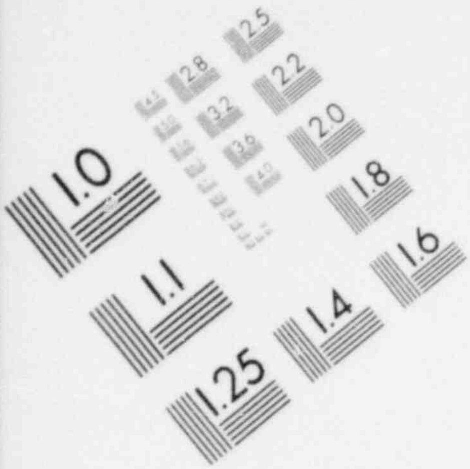
IMAGE EVALUATION
TEST TARGET (MT-3)



MICROCOPY RESOLUTION TEST CHART







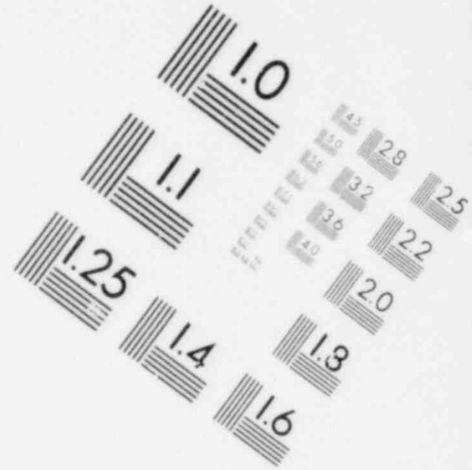
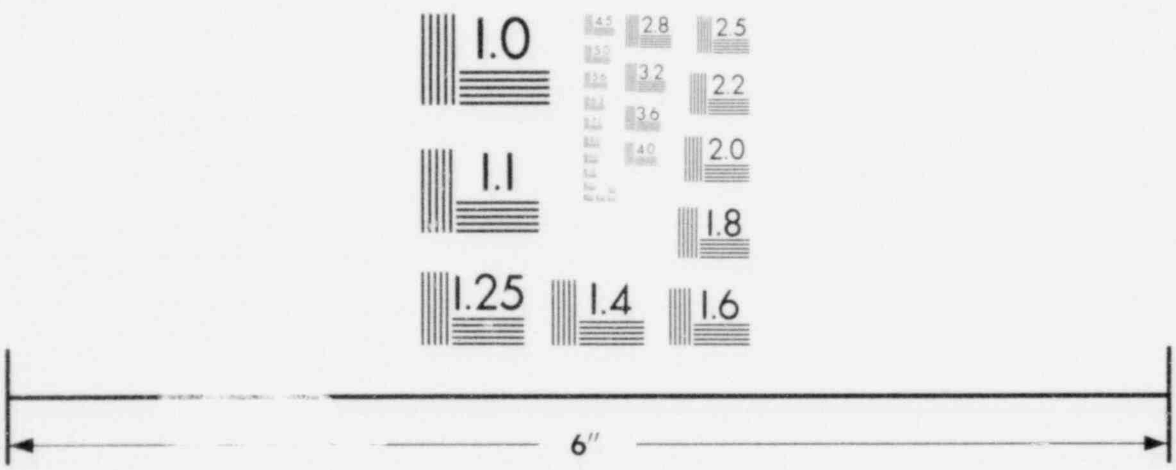
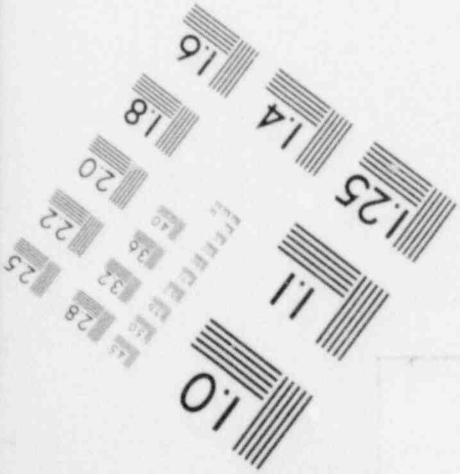


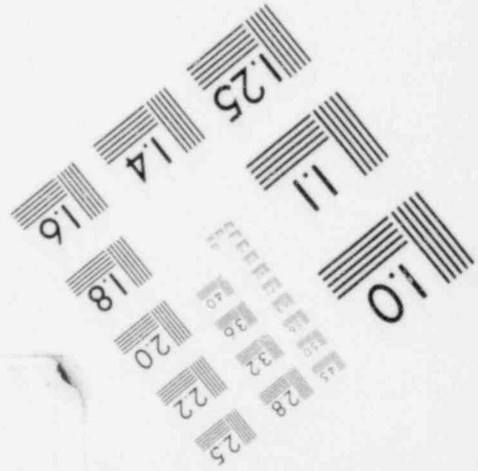
IMAGE EVALUATION TEST TARGET (MT-3)



6″

MICROCOPY RESOLUTION TEST CHART





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 approximates 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 is a nominal 2.62 weight percent of ^{235}U. Three fuel enrichments are used in the initial core. The highest enrichment is less than 3.5 weight percent ^{235}U.

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 3-59. The full-length CRA contain a 134-inch length of silver-indium-cadmium alloy clad with stainless steel. The APSRA contain a 36-inch length of silver-indium-cadmium alloy.(3)

5.3.1.5 The initial core has 68 burnable poison spider assemblies with similar dimensions as the full-length control rods. The cladding is Zircaloy-4 filled with alumina-boron and placed in the core as shown in FSAR Figure 3-2.

5.3.1.6 Reload fuel assemblies and rods shall conform to design and evaluation described in FSAR and shall not exceed an enrichment of 3.5 percent of ^{235}U.

5.3.2 Reactor Coolant System

5.3.2.1 The reactor coolant system is 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, are designed for a pressure of 2500 psig and a temperature of 650 F. The pressurizer and pressurizer surge line are designed for a temperature of 670 F.(5)

5.3.2.3 The reactor coolant system volume is less than 12,200 cubic feet.

REFERENCES

(1) FSAR, Section ~ 2.1

(2) FSAR, Section 3.2.2

(3) FSAR, Section 3.2.4.2

(4) FSAR, Section 4.1.2

5.4 NEW AND SPENT FUEL STORAGE FACILITIES

Applicability

Applies to storage facilities for new and spent fuel assemblies.

Objective

To assure that both new and spent fuel assemblies will be stored in such a manner that an inadvertent criticality could not occur.

Specification

5.4.1 New Fuel Storage

1. New fuel will normally be stored in the new fuel storage pool. The fuel assemblies are stored in racks in parallel rows, having a nominal center to center distance of 21 inches in both directions. This spacing is sufficient to maintain a Keff of less than .9 even if flooded with unborated water, based on fuel with an enrichment of 3.5 weight percent U235.

2. New fuel may also be stored in the spent fuel pool or in their shipping containers.

5.4.2 Spent Fuel Storage

1. Irradiated fuel assemblies will be stored, prior to offsite shipment, in the stainless steel lined spent fuel pool, which is located in the auxiliary building. The pool is sized to accommodate a full core of irradiated fuel assemblies in addition to the concurrent storage of the largest quantity of spent fuel assemblies predicted by the fuel management program.

2. The spent fuel pool is filled with borated water with a minimum concentration of 1800 ppm boron during refueling.

3. One spent fuel storage rack position is designed to accomodate a special container for storage of a leaking fuel assembly.

4. The spent fuel pool and new fuel pool racks are designed as seismic Class 1 equipment.

References

FSAR, Section 9.6

6. ADMINISTRATIVE CONTROLS

Applicability

Administrative controls are the means by which plant operations are subject to management control. Measures specified in this section provide for the assignment of responsibilities, plant organization, staffing qualifications and related requirements, review and audit mechanisms, procedural controls and reporting requirements. Each of the measures are applicable throughout plant life.

Objective

To ensure that adequate management controls are available for safe and efficient facility operation.

6.1 RESPONSIBILITY

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

6.1.2 In all matters pertaining to operation of the nuclear units and to these Technical Specifications, the Superintendent shall report to and be directly responsible to the Manager, Nuclear Services who reports to the Director, Power Production. The corporate management organization is shown in Figure 6.1-1.

6.1.3 In the absence of the Superintendent, the Assistant Superintendent will assume all responsibility and perform all duties of the Superintendent.

6.2 PLANT STAFF ORGANIZATION

The minimum functional organization for operation of the plant shall be as shown in Figure 6.2-1.

6.2.1 Minimum Shift Requirements

The minimum shift requirements for normal plant operations (whenever the plant is above cold shutdown) are as indicated in Table 6.2-1.

6.2.2 Special Plant Conditions

In addition to the requirements of 6.2.1, the following shift requirements shall be met:

6.2.2.1 One licensed operator shall be in the control room at all times when there is fuel in the reactor.

6.2.2.2 A licensed Senior Operator shall be on site at all times when there is fuel in the reactor.

6.2.2.3 Two licensed operators shall be in the control room during startup, normal shutdowns and during recovery from trips caused by transients or emergencies.

6.2.2.4 A licensed Senior Reactor Operator with no other concurrent operational duties shall directly supervise:

(A) All irradiated fuel handling and transfer activities onsite; and,

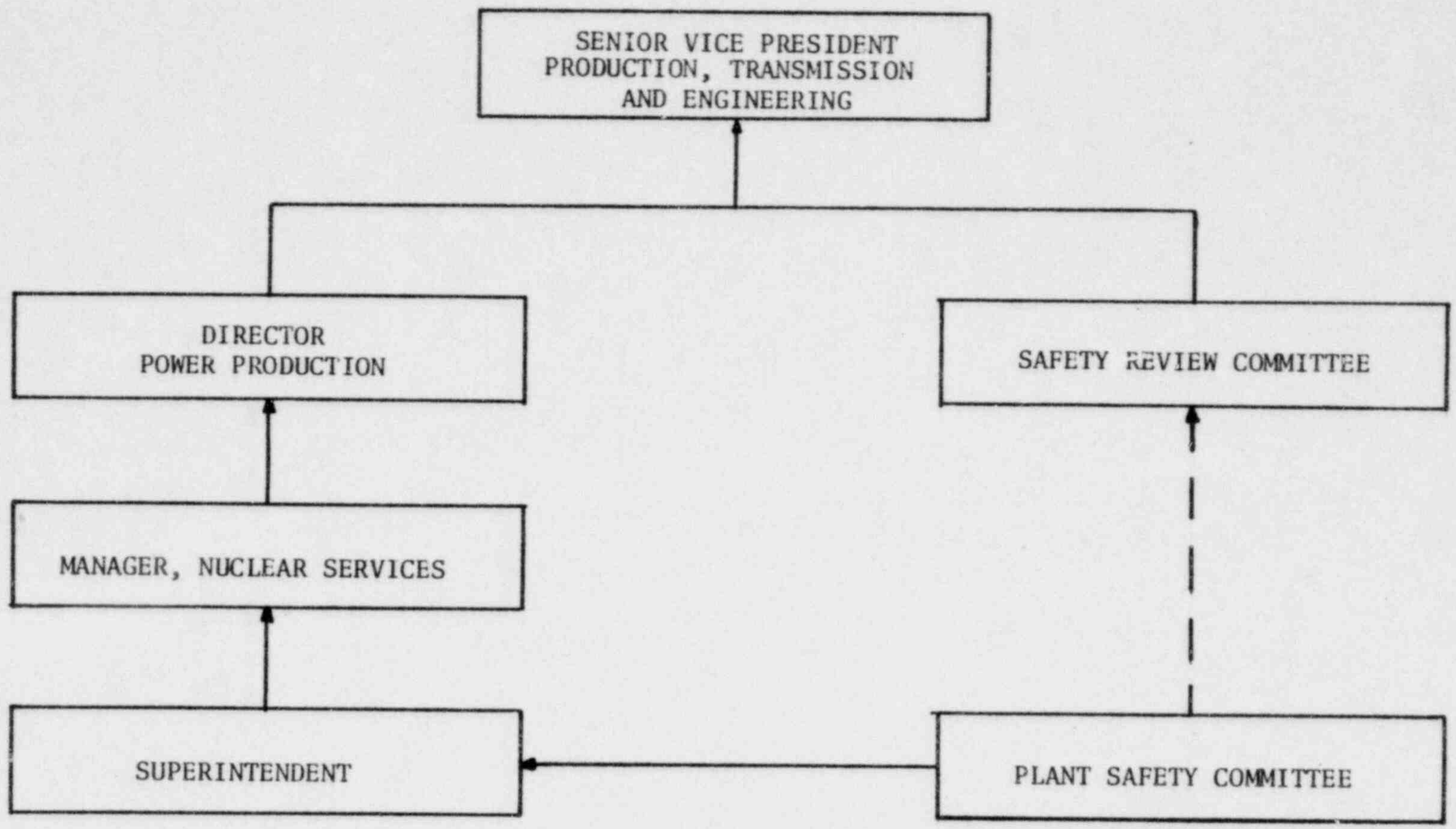
(B) All unirradiated fuel handling and transfer activities to and from the reactor vessel.

6.2.2.5 An individual who meets the qualifications of a health physics technician shall be on site at all times that there is fuel on site.

6.3 QUALIFICATIONS

Minimum qualifications, training, replacement training and retraining of plant personnel shall be in accordance with that stated in the Standard for Selection and Training of Personnel for Nuclear Power Plants, ANSI N18.1 - 1971. The minimum frequency of the retraining program shall be every two years. The training program shall be under the direction of a designated member of the plant staff.

Key personnel as designated on Figure 6.2-1 are those personnel whose qualifications must be reviewed by the AEC when they are appointed to the positions so designated.



ARKANSAS POWER & LIGHT CO. ARKANSAS NUCLEAR ONE-UNIT 1	MANAGEMENT ORGANIZATION CHART	FIGURE 6.1-1

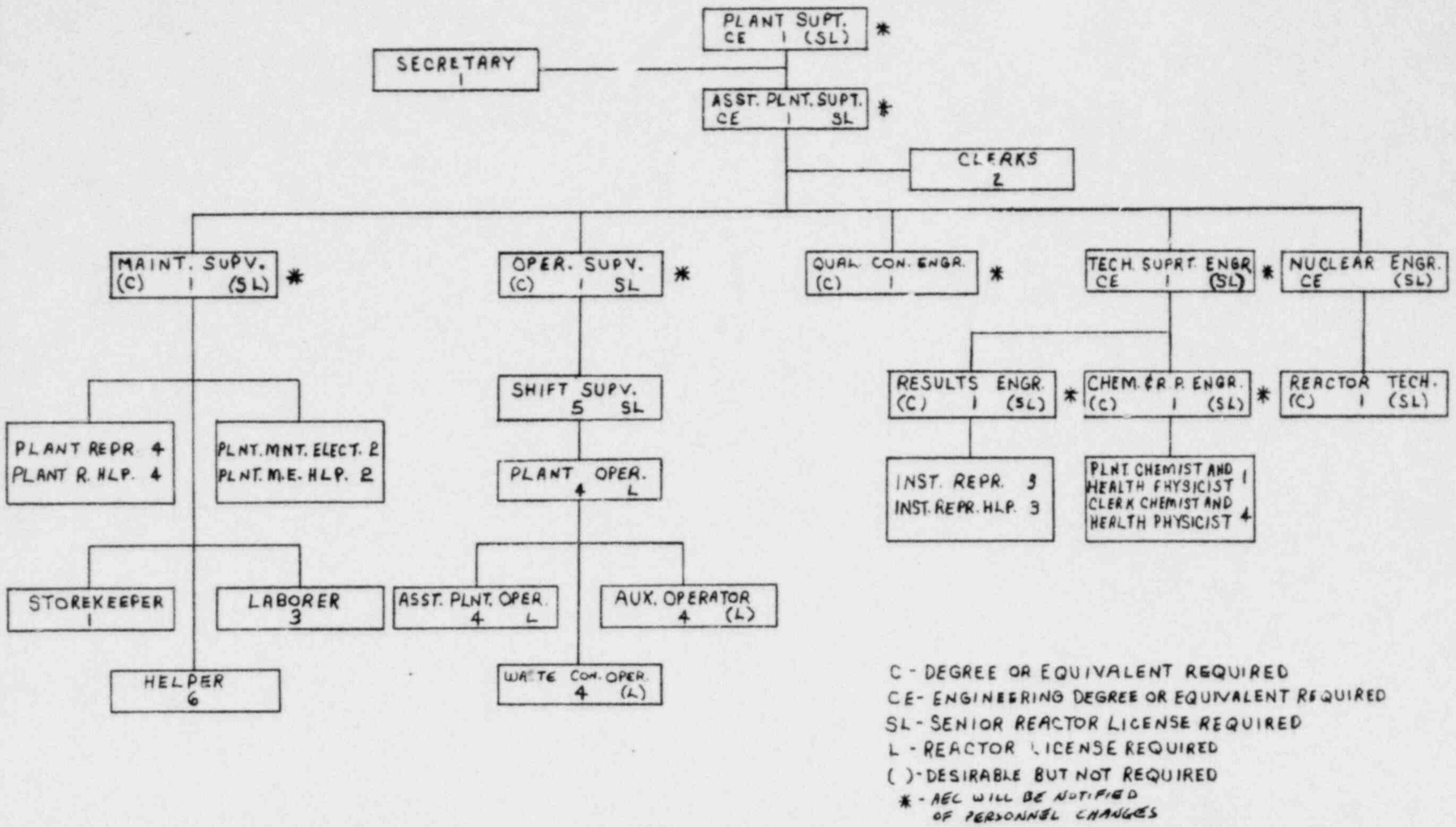
TABLE 6.2-1

ARKANSAS NUCLEAR ONE

MINIMUM OPERATING SHIFT REQUIREMENTS

UNIT 1

Shift Supervisor AEC Senior Reactor Operator License	1
Plant Operator AEC Reactor Operator License	1
Assistant Plant Operator AEC Reactor Operator License	1
Auxiliary Operator	1
Waste Control Operator	1
Men/Shift	5
Senior Reactor Operator License/Shift	1
Reactor Operator License/Shift	2



ARKANSAS POWER & LIGHT COMPANY ARKANSAS NUCLEAR ONE-UNIT 1	MINIMUM FUNCTIONAL ORGANIZATION FOR PLANT OPERATION	FIGURE 6.2-1

6.4 REVIEW AND AUDIT

Review and audit of station operation, maintenance, and technical matters shall be provided by two committees as follows:

6.4.1 Plant Safety Committee - A plant staff committee shall be constituted and function as described below:

a. Membership:

Assistant Superintendent (Chairman)
Technical Support Engineer
Chemical and Radiation Protection Engineer
Operations Supervisor
Maintenance Supervisor
Nuclear Engineer

The Superintendent shall appoint an acting chairman in the absence of the Assistant Superintendent.

b. Qualifications:

The qualifications of the regular members of the Plant Safety Committee with regard to experience and technical specialties of the individual members shall be maintained at a level at least equal to those described in ANSI N18.1 - 1971.

c. Consultants:

Additional personnel with expertise in specific areas such as radiochemistry, reactor engineering, and health physics may serve as consultants to the Plant Safety Committee.

d. Meeting Frequency:

Monthly, and as required, on call of the Chairman.

e. Quorum:

Chairman plus three members or their designated alternates.

f. Designated alternates shall be from other plant personnel in the appropriate disciplines; however, there shall be no more than two (2) alternate members serving on the committee at any one time.

g. Responsibilities:

1. The committee shall review proposed normal, off-normal and emergency operating procedures; proposed maintenance procedures and proposed changes thereto and any other proposed procedures or changes which affect nuclear safety.

2. The committee shall review proposed tests and experiments.

3. The committee shall review proposed changes to the Technical Specifications.

4. The committee shall review proposed changes or modifications to plant systems or equipment.

5. The committee shall review nuclear unit operations to detect any potential safety hazards.

6. The committee shall investigate reported instances of violations of the Technical Specifications, such investigations to include reporting, evaluation and recommendations to prevent recurrence, to the Superintendent.

7. The committee shall perform special reviews and investigations and render reports thereon as requested by the Superintendent.

8. The committee shall review the periodic drills to be conducted on emergency procedures, including evacuation (partial or complete) of the site and check adequacy of communications with off-site support groups.

9. The committee shall review the Industrial Security Plan and implementing procedures and shall submit recommended changes to the Superintendent.

10. The committee shall review the Emergency Plan and implementing procedures and shall submit recommended changes to the Superintendent.

h. Authority:

1. The Plant Safety Committee shall be advisory.

2. The Plant Safety Committee shall recommend to the Superintendent approval or disapproval of proposals under items g. (1) through (4) above.

   a. In the event of a disagreement between the recommendations of the Plant Safety Committee and the actions contemplated by the Superintendent, the course determined by the Superintendent to be more conservative will be followed. Records of the disagreement will be sent for review to the Manager, Nuclear Services, the Director, Power Production and the Safety Review Committee by the Superintendent.

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

i. Records:

Minutes shall be kept at the plant of all meetings of the Plant Safety Committee and copies shall be sent to the Manager, Nuclear Services, the Director Power Production and to the Chairman of the Safety Review Committee by the Superintendent.

j. Procedures:

Written administrative procedures for committee function shall be prepared and maintained describing: the method of submission and the content of presentations to the committee; provisions for the use of subcommittees; review and approval by members of written committee evaluations and recommendations; the distribution of minutes; and, such other matters as may be appropriate.

6.4.2 Safety Review Committee - A corporate committee shall be constituted and function as described below:

a. Membership:

Director Power Production (Chairman)
Manager, Nuclear Services
Maintenance and Operations Coordinator
Arkansas Power & Light Company Manager of Safety
Arkansas Nuclear One Superintendent
Chief Chemist
Arkansas Nuclear One Chemical & Radiation Protection Engineer
Arkansas Nuclear One Nuclear Engineer
Design and Construction Project Engineer
Radiation and Health Physics Consultant
Nuclear Safety Consultant

The Director, Power Production shall appoint an acting chairman in the absence of the Chairman. The plant staff members shall be non-voting members.

Alternates:

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

2. No more than 2 alternates shall serve on the committee at any one time.

3. Alternate members shall be non-voting.

b. Qualifications:

1. At least four voting members including participating alternates shall have a minimum of a Bachelor's Degree in Engineering or the Physical Sciences or equivalent, and have a minimum of three years of professional level experience, or the equivalent, in nuclear services, nuclear plant operation, or nuclear engineering, and the necessary overall nuclear background to determine when to call consultants and contractors for dealing with complex problems beyond the scope of the Company organization.

2. Members and alternates shall collectively have the capability required to review the areas of:

   a. reactor operations
   b. nuclear engineering
   c. chemistry and radiochemistry
   d. metallurgy
   e. instrumentation and control
   f. radiological safety
   g. mechanical and electrical engineering
   h. environmental considerations
   i. other appropriate fields required by the unique characteristics of the nuclear unit(s) involved.

   When the nature of a particular situation dictates, special consultants will be utilized to provide expert advice to board members upon request of any two board members.

3. If sufficient expertise in the specialty in b.2 above is not available from within the board, staff specialists and/or outside consultants shall be used to supplement review and audit functions of the board. Personnel in this category shall be competent in technical matters related to nuclear unit safety and other engineering and scientific support aspects.

c. Meeting Frequency:

The Safety Review Committee shall meet on call by the Chairman but not less frequently than twice a year. During the period of initial operation, Safety Review Committee shall be no less frequently than once per calendar quarter.

d. Quorum:

1. No less than a majority of the committee voting membership shall constitute a quorum.

2. Either the Chairman or Acting Chairman shall be present.

3. No more than a minority of the quorum shall have line responsibility for nuclear unit operation.

e. Purpose:

1. The Safety Review Committee shall conduct a critical examination of design, construction and those aspects of monitoring nuclear unit operation necessary to formulate an independent evaluation of contemplated actions, and after-the-fact investigations of anomalies.

2. The Safety Review Committee shall be constituted by a written charter stating:

   a. Subjects within purview of the board.
   b. Responsibility and authority.
   c. Mechanisms for convening meetings.
   d. Provisions for use of subgroups.
   e. Authority for access to unit records.
   f. Reporting requirements.

f. Authority and Responsibility:

The Safety Review Committee shall be advisory to the Senior Vice President, Production, Transmission and Engineering (PT&E) and AP&L corporate management.

g. Review and Audit:

The board will verify that: nuclear unit operation is consistent with Company policy, rules, approved operating procedures and license provisions; unusual events are promptly investigated and corrected in a manner which reduces the probability of recurrence of such events; trends are detected which may not be apparent to a day-to-day observer.

The operation of the nuclear unit(s) shall be formally audited on a periodic bases. These audits shall be performed no less frequently than semi-annually. Periodic review of the activity shall be performed by the board to assure that such audits are being accomplished in accordance with requirements of Technical Specifications. Such audits shall include verification of conformance with normal, off normal, maintenance and emergency surveillance, test and radiation control procedures and the Emergency and Security Plans. These audits shall be performed in accordance with appropriate written instructions or procedures and shall include verification of compliance with internal rules, procedures and regulations and license provisions, performance of the operating staff, and corrective actions following anomalies. Written reports of such audits shall be incorporated in the records of the board and disseminated to appropriate members of management, including those having responsibility in the area audited. Follow-up action, including re-audit of defici areas, shall be taken where indicated and the results r ed to responsible management levels on a formal ba

Subjects for review shall include:

1. Proposed tests and experiments, and results thereof, when these constitute an unreviewed safety question defined in 10 CFR 50.59.

2. Proposed changes in procedures, equipment, or systems which may involve an unreviewed safety question as defined in 10 CFR 50.59 (c) or changes which are referred by the Plant Superintendent. Also, new procedures which may affect nuclear safety.

3. Proposed Technical Specification or license changes.

4. Violations of statutes, regulations, orders, Technical Specifications, license requirements, or internal procedures or instructions having safety significance.

5. Significant operating abnormalities or deviations from normal performance of unit equipment whose failure would affect safe shutdown of the plant.

6. Abnormal occurrences as defined in Section 1.0.

7. The Emergency Plans and procedures.

8. The Industrial Security Plan and procedures.

9. Environmental monitoring.

10. Nuclear safety matters deemed essential to the safe operation of the facility by the Superintendent, the Plant Safety Committee, the Manager, Nuclear Services, the Director Power Production, or the Senior Vice President (PT&E).

11. Reports and meeting minutes of the Plant Safety Committee.

12. Reports submitted to the Atomic Energy Commission and associated responses.

h. Minutes.

Meeting minutes shall be prepared, formally approved, retained and promptly distributed to board members and other appropriate members of management having responsibility in the areas reviewed.

6.5 ACTION TO BE TAKEN IN THE EVENT OF AN ABNORMAL OCCURRENCE

6.5.1 Any abnormal occurrence shall be reported immediately to the Manager, Nuclear Services, the Director Power Production, and the Senior Vice President (PT&E) and promptly reviewed by the Plant Safety Committee.

6.5.2 The Plant Safety Committee shall prepare a separate report for each abnormal occurrence. This report shall include an evaluation of the cause of the occurrence, a record of the corrective action taken, and recommendations for appropriate action to prevent or reduce the probability of a recurrence.

6.5.3 Copies of all such reports shall be submitted to the Superintendent for distribution to the Manager, Nuclear Services, the Director Power Production, the Senior Vice President (PT&E), and to the Chairman of the Safety Review Committee for review and approval of any recommendations.

6.5.4 The Senior Vice President (PT&E) shall report the circumstances of any abnormal occurrence to the AEC as specified in Section 6.12, "Plant Reporting Requirements".

6.6 ACTION TO BE TAKEN IF A SAFETY LIMIT IS EXCEEDED

6.6.1 If a safety limit is exceeded, the reactor shall be shut down and reactor operation shall only be resumed in accordance with the provisions of 10 CFR 50.36 (c) (1) (i).

6.6.2 An immediate report shall be made to the Manager, Nuclear Services, the Director Power Production, the Senior Vice President (PT&E), and the Chairman of the Safety Review Committee.

6.6.3 The Senior Vice President (PT&E) shall promptly report the circumstances to the AEC as specified in Section 6.12, "Plant Reporting Requirements".

6.6.4 A complete investigation of the occurrence including an analysis of the circumstances leading up to and resulting from the occurrence together with recommendations to prevent a recurrence shall be prepared by the Plant Safety Committee. This report shall be submitted to the Manager, Nuclear Services, Director Power Production, the Senior Vice President (PT&E) and the Chairman of the Safety Review Committee by the Superintendent. Appropriate analyses or reports will be submitted to the AEC by the Senior Vice President (PT&E) as specified in Section 6.12, "Plant Reporting Requirements".

6.7 PLANT OPERATING PROCEDURES

6.7.1 Detailed written procedures, covering areas listed below, shall be prepared, approved as specified in Section 6.7.2, and adhered to for operation of all systems and components involving nuclear safety.

a. Normal startup, operation and shutdown of the reactor and of all systems and components involving nuclear safety of the facility.

b. Refueling operations.

c. Actions to be taken to correct specific and foreseen potential malfunctions of systems or components, including responses to alarms, suspected primary system leaks and abnormal reactivity changes.

d. Emergency conditions involving potential or actual release of radioactivity.

e. Preventive or corrective maintenance operations involving nuclear safety of the facility.

f. Emergency and off-normal condition procedures.

g. Surveillance and testing requirements.

h. Plant Security Plan implementation.

i. Emergency Plans implementation.

6.7.2 All procedures described in 6.7.1 above, and changes thereto, shall be reviewed by the Plant Safety Committee and approved by the Superintendent prior to implementation, except as provided in 6.7.3 below.

6.7.3 Temporary changes to procedures in 6.7.1 above, which do not change the intent of the original procedure may be made, provided such changes are approved by two members of the plant staff, at least one of whom shall be a Shift Supervisor. Temporary changes which may affect the intent of the original procedure may be made, provided such changes are approved by the Plant Superintendent.

6.8 RADIATION AND RESPIRATORY PROTECTION PROGRAM

6.8.1 Radiation control procedures shall be provided and made available to all plant personnel. These procedures will show the permissible radiation exposure. This radiation protection program will be organized to meet the requirements of 10 CFR 20, with the following exceptions:

1. Pursuant to 10 CFR 20.103 (c) (1) and (3), allowance can be made for the use of respiratory protective equipment in conjunction with activities authorized by the operating license for this plant in determining whether individuals in restricted areas are exposed to concentrations in excess of the limits specified in Appendix B, Table I, Column 1, of 10 CFR 20, subject to the following conditions and limitations:

   a. The limits provided in Section 20.103 (a) and (b) are not exceeded.

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

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

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

   a. The limits specified in paragraph 1 of this section are not exceeded.

   b. Respiratory protective equipment is selected and used so that the peak concentrations of airborne radioactive material inhaled by an individual wearing the equipment does not exceed the pertinent concentration values specified in Appendix B, Table I, Column 1, of 10 CFR 20. For the purposes of this subparagraph, the concentration of radioactive material that is inhaled when respirators are worn may be determined by dividing the ambient airborne concentration by the protection factor specified in Table 6.8-1, appended to this specification, for the respirator protective equipment worn. If the intake of radioactivity is later determined by other measurements to have been different than that initially estimated, the latter quantity shall be used in evaluating the exposures.

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

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

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

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

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

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

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

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

e. The licensee uses equipment approved by the U. S. Bureau of Mines under its appropriate Approval Schedules as set forth in Table 6.8-1 below. Equipment not approved under U. S. Bureau of Mines Approval Schedules may be used only if the licensee has evaluated the equipment and can demonstrate by testing, or on the basis of reliable test information, that the material and performance characteristics of the equipment are at least equal to those afforded by U. S. Bureau of Mines approved equipment of the same type, as specified in Table 6.8-1 below.

f. Unless otherwise authorized by the Commission, the licensee does not assign protection factors in excess of those specified in Table 6.8-1 below in selecting and using respiratory protective equipment.

3. These specifications with respect to the provisions of 20.103 shall be superseded by adoption of proposed changes to 10 CFR 20.103, which would make this specification unnecessary.

TABLE 6.8-1
PROTECTION FACTORS FOR RESPIRATORS

DESCRIPTION	MODES	PROTECTION FACTORS 2/ PARTICULATES AND VAPORS AND GASES EXCEPT TRITIUM OXIDE 3/	GUIDES TO SELECTION OF EQUIPMENT BUREAU OF MINFS APPROVAL SCHEDULES* FOR EQUIPMENT CAPABLE OF PROVIDING AT LEAST EQUIVALENT PROTECTION FACTORS *or schedule superseding for equipment of type listed
I. AIR-PURIFYING RESPIRATORS			
Facepiece, half-mask 4/ 7/	NP	5	21B 30 CFR 14.4(b) (4)
Facepiece, full 7/	NP	100	21B 30 CFR 14.4(b) (5); 14F 30 CFR 13
II. ATMOSPHERE-SUPPLYING RESPIRATOR			
1. Airline Respirator			
Facepiece, half-mask	CF	100	19B 30 CFR 12.2(c) (2) Type C(i)
Facepiece, full	CF	1,000	19B 30 CFR 12.2(c) (2) Type C(i)
Facepiece, full 7/	D	500	19B 30 CFR 12.2(c) (2) Type C(ii)
Facepiece, full	PD	1,000	19B 30 CFR 12.2(c) (2) Type C(iii)
Hood	CF	5/	6/
Suit	CF	5/	6/
2. Self-Contained Breathing Apparatus (SCBA)			
Facepiece, full 7/	D	500	13E 30 CFR 11.4(b) (2) (i)
Facepiece, full	PD	1,000	13E 30 CFR 11.4(b) (2) (ii)
Facepiece, full	R	1,000	13E 30 CFR 11.4(b) (1)
III. COMBINATION RESPIRATOR Any combination of air-purifying and atmosphere-supplying respirator		Protection factor for type and mode of operation as listed above	19B CFR 12.2(e) or applicable schedules as listed above.

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

TABLE 6.8-1 NOTES

1/ See the following symbols:

CF: Continuous Flow
D: Demand
NP: Negative Pressure (i.e., negative phase during inhalation)
PD: Pressure Demand (i.e., always positive pressure)
R: Recirculating (closed circuit)

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

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

(b) The protection factors apply:

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

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

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

3/ Excluding radioactive contaminants that present an absorption or submersion hazard. For tritium oxide approximately half of the intake occurs by absorption through the skin so that an overall protection factor of not more than approximately 2 is appropriate when atmosphere-supplying respirators are used to protect against tritium oxide. Air-purifying respirators are not recommended for use against tritium oxide. See also footnote 5/, below, concerning supplied-air suits and hoods.

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

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

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

7/ Only for shaven faces.

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

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

6.9 EMERGENCY PLANNING

6.9.1 An Emergency Plan shall be maintained throughout the life of the plant. This plan shall be reviewed annually.

6.9.2 Interfaces with Federal, State or local agencies required for effective implementation of the Emergency Plan shall be maintained.

6.9.3 Emergency drills shall be conducted at least annually.

6.10 INDUSTRIAL SECURITY PROGRAM

6.10.1 An industrial security program shall be maintained throughout the life of the plant in accordance with the provisions of the ANO Industrial Security Plan. Annual review of the Plant Security Plan will be performed.

6.10.2 Investigations of all attempted or actual security infractions shall be conducted by the Superintendent, in cooperation with any Federal, State, or local agencies involved, and a report filed with the Manager, Nuclear Services, the Director Power Production, Senior Vice President (PT&E), and Chairman of the Safety Review Committee.

6.10.3 Industrial Security violations shall be reported as indicated in Specification 6.12.

6.11 RECORDS RETENTION

6.11.1 All records and logs relative to the following areas shall be retained for 5 years:

a. Records of normal nuclear unit operation, including power levels and periods of operation at each power level.

b. Records of principal maintenance activities, including inspection, repair, substitution or replacement of principal items of equipment pertaining to nuclear safety.

c. Records of abnormal occurrences.

d. Records of periodic checks, inspections and calibrations performed to verify that surveillance requirements are being met.

e. Records of any special reactor test or experiments.

f. Records of changes made in the Operating Procedures.

g. Records of radioactive shipments.

h. Test results, in units of microcuries, for leak tests performed pursuant to Specification 4.14.

i. Record of annual physical inventory verifying accountability of sources on record.

6.11.2 All records relative to the following areas shall be retained for the life of the plant:

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

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

c. Records of plant radiation and contamination surveys.

d. Records of off-site environmental monitoring surveys.

e. Records of radiation exposure of all plant personnel, and others who enter radiation control areas.

f. Records of radioactivity in liquid and gaseous wastes related to the environment.

g. Records of transient or operational cycling for those plant components that have been designed to operate safely for a limited number of transients or operational cycles.

h. Records of current individual plant staff members indicating qualifications, experience, training and retraining.

i. Reactor coolant system inservice inspections.

j. Minutes of meeting of the Safety Review Committee.

6.12 PLANT REPORTING REQUIREMENTS

6.12.1 The following information shall be submitted in addition to the reports required by Title 10, Code of Federal Regulations.

6.12.2 Routine Reports:

Operations Reports shall be submitted in writing to the Director, Regulatory Operations, Region II, USAEC, Atlanta, Georgia.

6.12.2.1 Startup Report

A summary report of unit startup and power escalation testing and an evaluation of the results from these test programs shall be submitted following receipt of operating licenses, following amendments to the licenses involving a planned increase in power level, following the installation of a new core or following modifications to an extent that the nuclear, thermal or hydraulic performance of the unit may be significantly altered. The test results shall be compared with design predictions and specifications. Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation (i.e., following synchronization of the turbo-generator to produce commercial power and turn over to the System Dispatcher) or (3) 9 months following initial criticality whichever is earliest.

6.12.2.2 First Year Operation Report

A report shall be submitted within 60 days after completion of the first year of commercial power operation as defined above. This report may be incorporated into the semiannual operating report and shall cover the following:

(a) an evaluation of unit performance to date in comparison with design predictions and specifications;

(b) a reassessment of the safety analysis submitted with the license application in light of measured operating characteristics when such measurements indicate that there may be substantial variance from prior analysis;

(c) an assessment of the performance of structures, systems and components important to safety;

(d) a progress and status report on any items identified as requiring additional information during the operating license review or during the startup of the plant, including items discussed in the AEC's safety evaluation, items on which additional information was required as conditions of the license and items identified in the licensee's startup report.

6.12.2.3 Semiannual Operating Reports

Semiannual operating reports covering the preceding six months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The first such period should begin with the date of initial criticality. These reports should include the following:

(a) Operations Summary

A summary of operating experience occurring during the reporting period that relates to the safe operation of the plant, including a summary of:

(1) changes in plant design,

(2) performance characteristics (e.g., equipment and fuel performance),

(3) changes in procedures which were necessitated by (1) and (2) or which otherwise were required to improve the safety of facility operations,

(4) results of surveillance tests and inspections required by these technical specifications,

(5) the results of any periodic containment leak rate tests performed during the reporting period.

(6) a brief summary of those changes, tests and experiments requiring authorization from the Commission pursuant to 10 CFR 50.59(a),

(7) any changes in the plant operating organization which involve positions which are designated as key supervisory personnel on Figure 6.2-1, and

(8) results of required leak tests performed on sources if the tests reveal the presence of 0.005 uCi or more of removable contamination.

(b) Power Generation

A summary of power generated during the reporting period including:

(1) gross thermal power generated (in MWH)

(2) gross electrical power generated (in MWH)

(3) net electrical power generated (in MWH)

(4) number of hours the reactor was critical

(5) number of hours the generator was on-line

(6) histogram of thermal power vs. time

(c) Shutdowns

Descriptive material covering all outages occurring during the reporting period. For each outage, information shall be provided on:

(1) the cause of the outage,

(2) the method of shutting down the reactor; e.g., trip, automatic rundown, or manually controlled deliberate shutdown,

(3) duration of the outage,

(4) unit status during the outage; e.g., cold shutdown or hot shutdown,

(5) corrective action taken to prevent repetition, if appropriate.

(d) Maintenance

A discussion of safety-related maintenance (excluding preventative maintenance) performed during the reporting period on systems and components that are designated to prevent or mitigate the consequences of postulated accidents or to prevent the release of significant amounts of radioactive material. Included in this category are systems and components which are part of the reactor coolant pressure boundary defined in 10 CFR 50.2(v), any part of the engineered safety features, or associated service and control systems that are required for the normal operation of engineered safety features, part of any

any reactor protection or shu down systems, or part of any radioactive waste treatment handling and disposal system or other system which may contain significant amounts of radioactive material. For any malfunctions for which corrective maintenance was required, information shall be provided on:

(1) the system or component involved,

(2) the cause of the malfunction

(3) the results and effects on safe operation,

(4) corrective action taken to prevent repetition,

(5) precautions taken to provide for reactor safety during repair.

(e) Changes, Tests and Experiments

A summary of all changes in the plant design and procedures that relate to the safe operation of the plant shall be included in the Operations Summary section of these semiannual reports. Changes, tests, and experiments performed during the reporting period that require authorization from the Commission pursuant to 10 CFR 50.59(a) are covered in paragraph 6.12.4 of these technical specifications; however, those changes, tests, and experiments that do not require Commission authorization pursuant to §50.59(a) shall be addressed. The report shall include a brief description and summary of the safety evaluation for those changes, tests, and experiments, carried out without prior Commission approval, pursuant to the requirements of §50.59(b) of the Commission's regulations, that "The licensee shall furnish to the Commission, annually or at such shorter intervals as may be specified in the license, a report containing a brief description of such changes, tests, and experiments, including a summary of the safety evaluation of each".

(f) Radioactive Effluent Releases & Environmental Monitoring

Reporting requirements for radioactive effluent releases and environmental monitoring program results are described in the Environmental Technical Specifications (Appendix B to the Operating License).

g. FSAR Changes

Revised FSAR pages shall be submitted on a replacement page basis appropriately prepared for direct insertion into the applicable FSAR section and describing any safety-related changes in facility design, method of operation, revised safety or transient analysis, or facility equipment additions. Also a listing of effective pages by date of revision or revision number shall be submitted.

6.12.3 Non-Routine Reports

6.12.3.1 Reporting of Abnormal Events

Notification shall be made within 24 hours by telephone and telegraph to the Director of the Regional Regulatory Operations Office, followed by a written report within 10 days to the Director, Directorate of Licensing (cc. to the Director of the Regional Regulatory Operations Office) in the event of the abnormal occurrences as defined in Section 1.0. The written report on these abnormal occurrences, and to the extent possible, the preliminary telephone and telegraph notification shall: (a) describe, analyze and evaluate safety implications, (b) outline the measures taken to assure that the cause of the condition is determined and (c) indicate the corrective action (including any changes made to the procedures and to the quality assureance program) taken to prevent repetition of the occurrence and of similar occurrences involving similar components or systems.

In addition, the written report shall relate any failures or degraded performance of systems and components for the incident to similar equipment failures that may have previously occurred at the plant. The evaluation of the safety implications of the incident should consider the cumulative experience obtained from the record of previous failures and malfunctions of the affected systems and components or of similar equipment.

6.12.3.2 Reporting of Unusual Events

A written report shall be forwarded within 30 days to the Director, Directorate of Licensing and to the Director of the Regional Regulatory Operations Office, in the event of:

a. Discovery of any substantial errors 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.

b. Any substantial variance from performance specifications contained in the technical specifications or in the Safety Analysis Report.

c. Any condition involving a possible single failure which, for a system designed against assumed single failures, could result in a loss of the capability of the system to perform its safety function.

6.12.4 Special Reports

Special reports shall be submitted in writing within 90 days to the Director, Directorate of Licensing, USAEC, Washington, D.C. 20545.

Special reports shall be submitted covering inspections, tests and maintenance that are appropriate to assure safe operation of the plant. The frequency and content of these special reports are determined on an individual case basis and designated in these technical specifications. Examples of subjects for such reports include:

(a) In-service inspection.

(b) Tendon surveillance

(c) Containment structural tests.

(d) Special maintenance reports.

(e) Authorization of changes, tests, and experiments in accordance with 10 CFR 50.59.

(f) Containment leak rate tests.

6.12.5 Personnel Exposure and Monitoring Reports

Routine reports of personnel exposure to ionizing radiation will be made to the U. S. Atomic Energy Commission as required by 10 CFR 20.407. Other notifications and reports will be made as required by 10 CFR 20.405, 20.408 and 20.409.